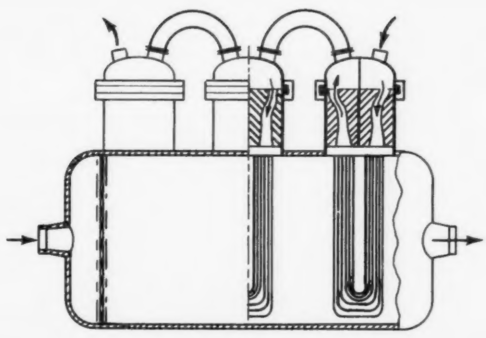
Journal Name Will Be Changed!
See Page ii.

REACTOR
AND FUEL-PROCESSING
TECHNOLOGY

A Quarterly Technical Progress Review

Prepared for DIVISION OF TECHNICAL INFORMATION, U. S. ATOMIC ENERGY COMMISSION



Heat Exchanger for the BN-350 (See Page 330)

Fall 1969

VOLUME 12

NUMBER 4

TECHNICAL PROGRESS REVIEWS

The United States Atomic Energy Commission publishes the Technical Progress Reviews to meet the needs of industry and government for concise summaries of current nuclear developments. Each journal digests and evaluates the latest findings in a specific area of nuclear technology and science. *Nuclear Safety* is a bimonthly journal; the other three are quarterly journals.

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REACTOR
AND FUEL-PROCESSING
TECHNOLOGY

Vol. 12, No. 4 Fall 1969

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New Name, Cover, and Price

New Name. Beginning with Volume 13, Number 1 (Winter 1969–1970), the name of this journal will be shortened to *Reactor Technology*. The new name will better reflect the contents now that fuel processing will no longer be a major part of the contents.

New Cover. Also beginning with the next issue, the cover we have used for the past 12 years will be replaced by a new design. We plan to make the cover of each issue distinctive in appearance by the use of an appropriate photograph or a drawing pertaining to the contents.

New Price. Effective with this issue (Vol. 12, No. 4), the price of this journal is $3.00 per year (four issues) or $0.75 per issue. (An additional amount is required for foreign mailing; see the back cover for complete ordering instructions.)

Issued quarterly by the U. S. Atomic Energy Commission. Use of funds for printing this publication approved by the Director of the Bureau of the Budget on May 12, 1969.

Seismic Analysis of Primary Piping Systems for Nuclear Generating Stations

By Lawrence Berkowitz*

Abstract: *A description of what we consider to be an adequate seismic analysis of primary piping systems, including major components such as steam generators and primary coolant pumps, is presented. Examples of how we would approach the analysis of typical, large, complex systems by lumped-mass mathematical model representations are given. Dynamic coupling of the piping system with shielding or containment structure is considered. The advantages and disadvantages of response-spectra, time-dependent, and spectral-density inputs are discussed. A step-by-step procedure for determining the seismic response to response-spectra inputs is summarized. The theoretical basis for computing the response to translatory as well as rocking motion of the base is given. Also, an example of computer results for response-spectra input is given.*

In recent years the designer of nuclear power plants has been required to demonstrate the earthquake resistance of proposed containment-structure and equipment designs. The first nuclear power-plant seismic loadings were specified by an equivalent static g load in the horizontal and vertical directions. Plants of more recent design have had seismic loadings specified in the form of frequency-dependent "response-spectra" curves or in the form of actual or artificial time-dependent ground motions. In addition, a consideration of soil-structure interaction gives rise to the possibility of base rocking motion as well as translatory motions.

General discussions on structural dynamic problems in the seismic design of nuclear power plants may be found in Refs. 1 to 3. This discussion concerns itself with the seismic analysis of primary piping systems for nuclear generating stations as might be typically associated with pressurized-water reactors. The dynamic analysis of piping systems has been characterized as being "very involved."[2] Also, according to Ref. 4, "an exact dynamic analysis of a complete piping system would present great difficulties" because a large number of concentrated masses are required to represent the pipe system, because the elastodynamic properties of supporting systems have to be accounted for, and because various parts of the nuclear power plant oscillate in different magnitudes and directions. It has therefore been suggested that dynamic analysis of piping systems be replaced whenever possible by a static analysis using high seismic coefficients. We believe, however, that it is possible to perform a meaningful, dynamic seismic analysis of Class I† primary piping systems, provided that the analyst has at his disposal adequate computer programs combined with an ability to exercise good engineering judgment in developing appropriate mathematical models.

DESCRIPTION OF TYPICAL PRIMARY PIPING SYSTEMS

An example of a primary piping system is shown schematically in Figs. 1 and 2. Note that this nuclear steam supply system consists of the following:

1. Reactor.
2. Two steam generators.
3. Two hot legs.
4. Four pumps.
5. Four cold legs from steam generator to pumps.
6. Four cold legs from pumps to reactor.
7. Pressurizer.

*The Franklin Institute Research Laboratories, Philadelphia, Pa.

†The proposed draft of USAS B31.7, Nuclear Power Piping, defines Class I piping as "those piping systems whose loss or failure could cause or increase the severity of a nuclear incident."

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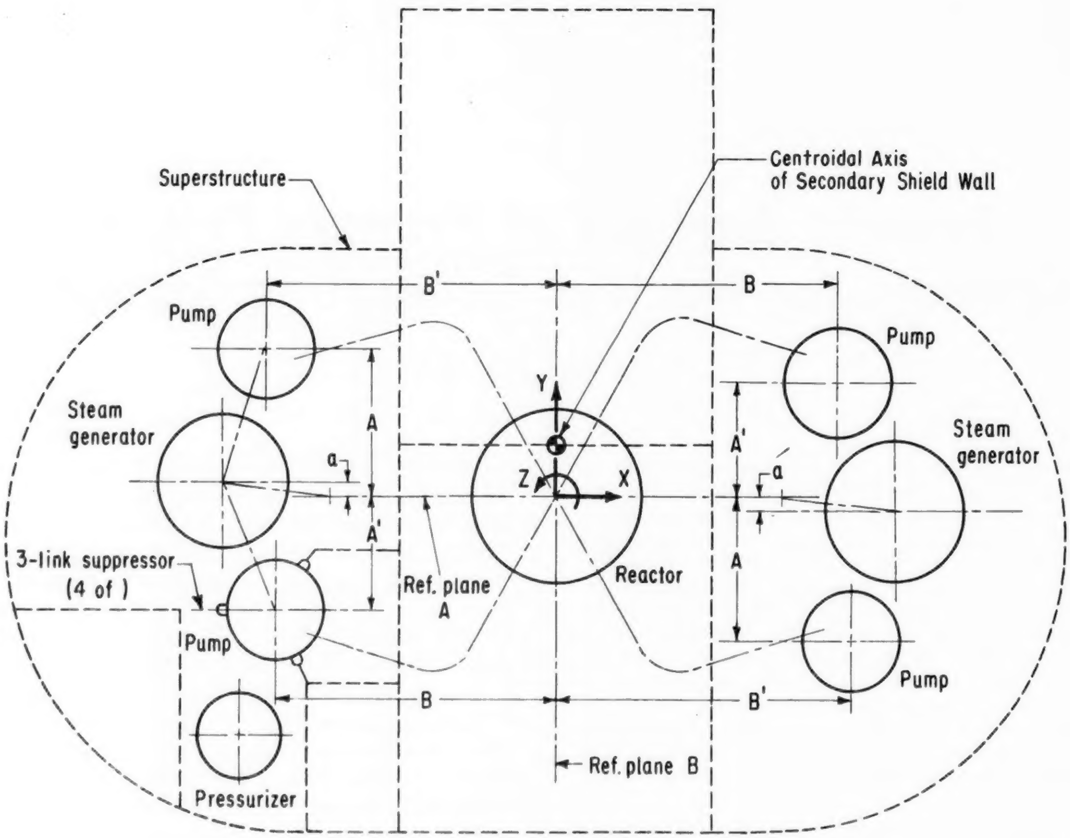


Fig. 1 Plan view of nuclear steam supply system Model A.

The plan-view representation of the system is shown in Fig. 1. Note that from what at first glance appears to be an arrangement that is symmetric about reference planes A and B is in fact not, although the lack of symmetry about plane B is less than about plane A.

Note that the four primary coolant pumps are connected to the secondary shield wall by three-link suppressors. The suppressors are designed to be flexible under statically applied loads (thus allowing thermal growth) but become stiff under dynamic loads that might occur during an earthquake. Accordingly, the system is coupled to the wall under seismic loading.

Figure 2 shows a perspective view of one loop of the system. Note that in this system kinematic constraints in the form of pipe supports, anchors, and hangers are minimal. Only the pumps are laterally braced by the suppressors. Note that the pumps in this system are supported by hangers from above.

An example of a system that does have piping constraints is the primary loop shown in Fig. 3. Note that the pumps in this system are supported from below. The loop shown in Fig. 3 had been anchored to the reactor and the containment building.

Figure 4 is an isometric drawing of another model of this loop. In this model the primary coolant pump and steam generator are assumed to be dynamically coupled to the containment building. In addition, the pump impeller shaft and housing, together with bearing supports, have been represented.

MODELING FOR SEISMIC RESPONSE

The analyst is faced with the initial problem of idealizing the piping system with a mathematical model that will reveal the significant modes of the structure. It has been our experience that the primary piping system, including its components, can adequately be

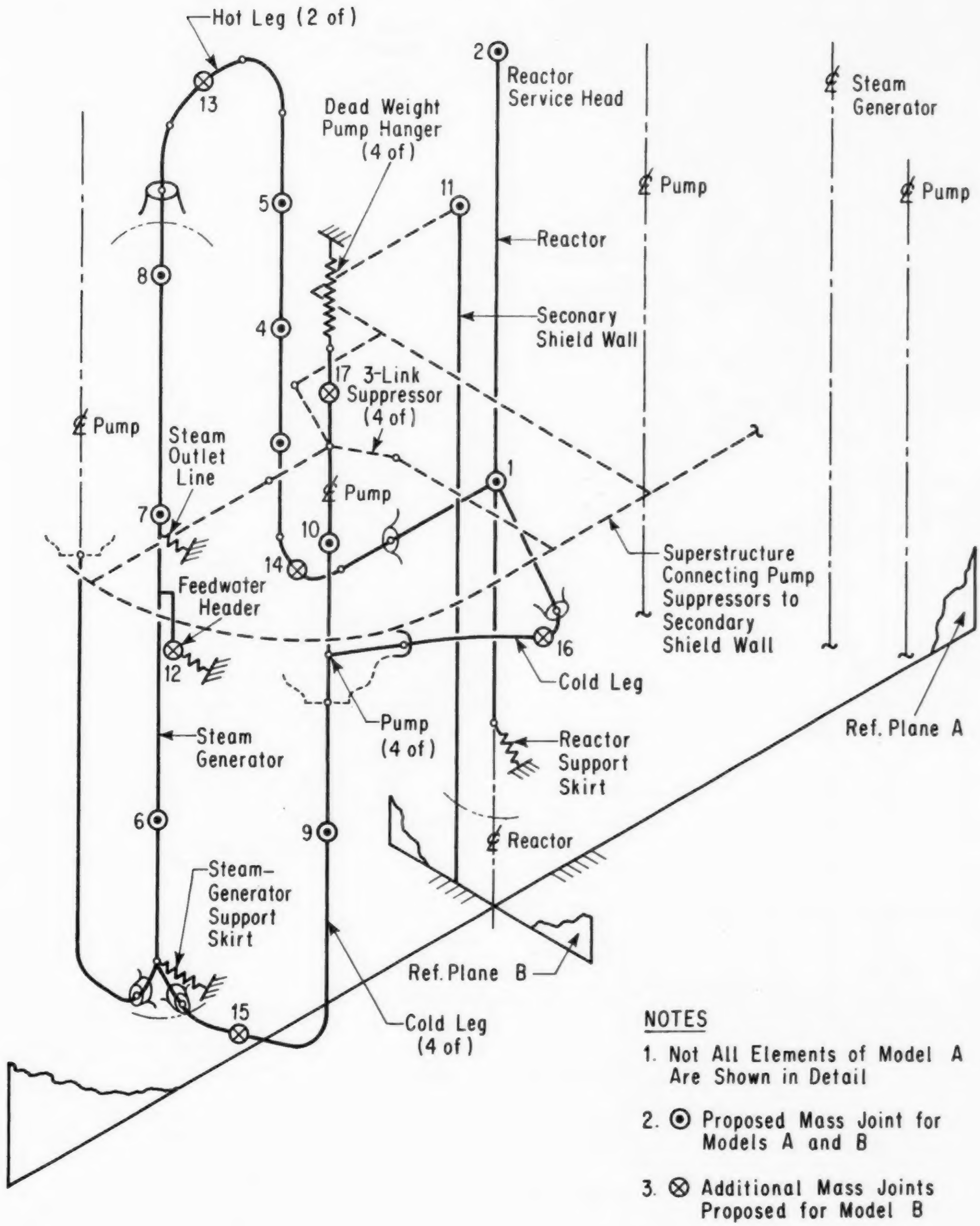


Fig. 2 Nuclear steam supply system.

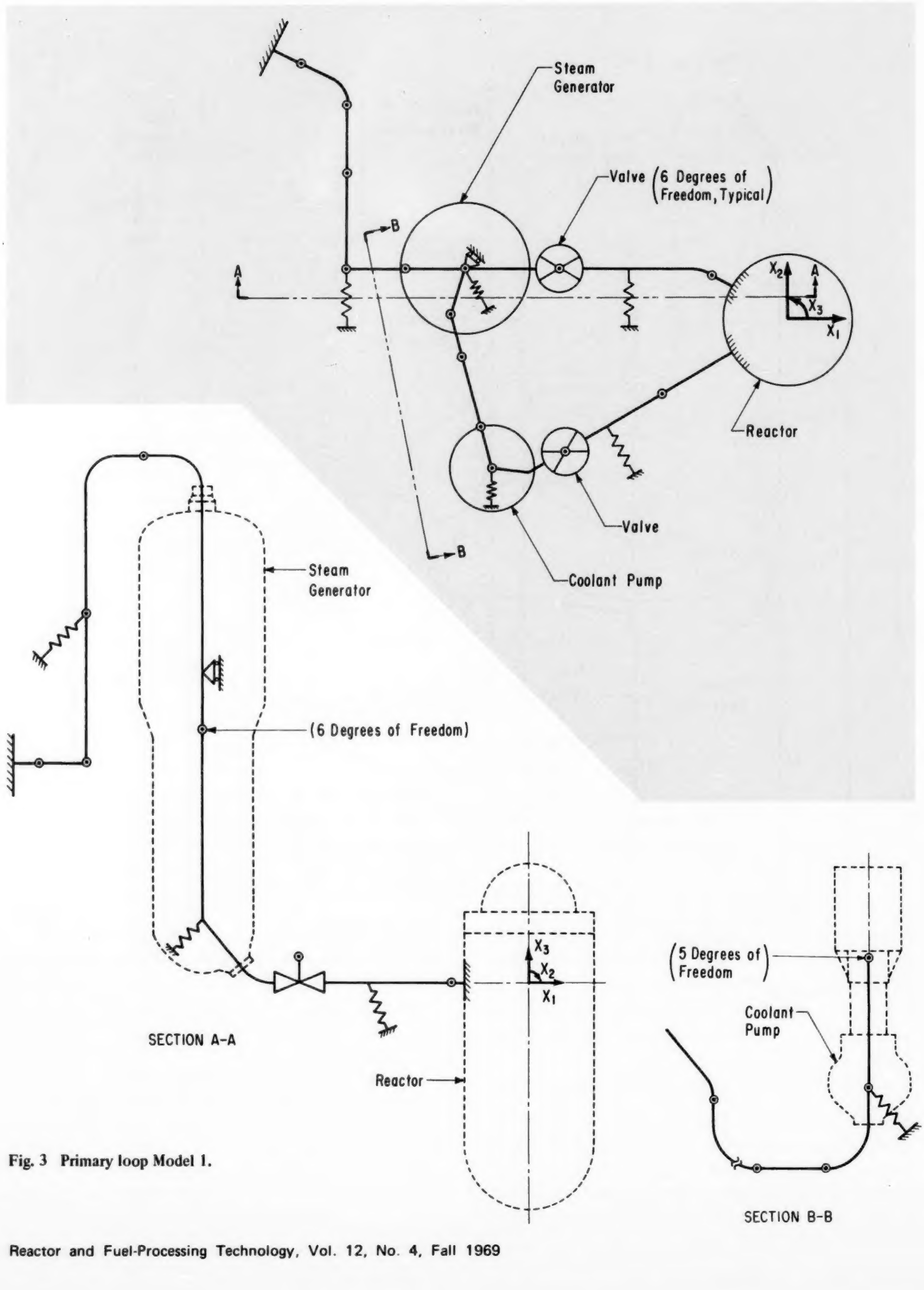


Fig. 3 Primary loop Model 1.

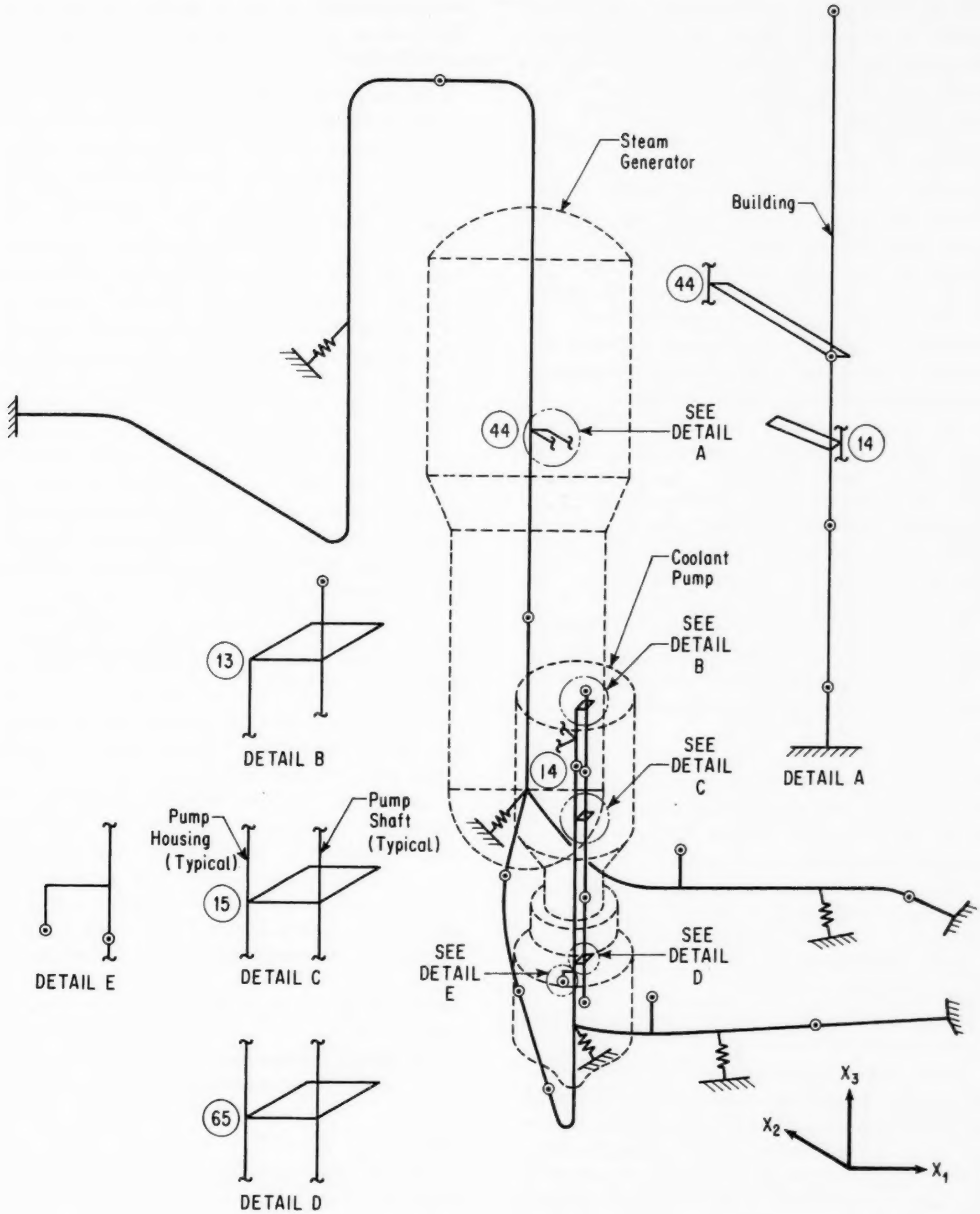


Fig. 4 Primary loop Model 2.

represented by a three-dimensional assemblage of beam elements with straight or curved centroidal axes and of uniform or nonuniform cross section, connected at structural joints with mass characteristics (translatory as well as rotational) concentrated at selected structural joints designated "mass joints."

Note that, although the analyst is primarily interested in the response of the piping system, the possibility of dynamic coupling with the containment structure such as the secondary shield wall shown in Fig. 1 should not be neglected. The question as to whether or not dynamic interaction between the walls and the pump should be considered can be answered by first assuming that the walls and pump are decoupled (neglecting the kinematic constraint imposed on the pump by the suppressors) and examining the mode shapes of the decoupled systems.

The natural frequency, ω_p, associated with the mode shape that exhibits a relatively large displacement of the pump should be compared with the fundamental frequency, ω_0, of the wall. If ω_p is considerably less than ω_0, then it is sufficient to neglect dynamic interaction between the wall and the loop, but to retain the kinematic constraint on the pump imposed by the lateral bracing.

If ω_p is close in value to ω_0, then the wall and steam supply systems should be dynamically coupled. In this case it may be sufficient to model the wall as a one mass system such that the fundamental frequency, ω_0, is retained.

The mathematical model of the piping systems should be capable of revealing the response to the anticipated dynamic inputs. The model should, of course, be compatible with the computer programs available to the analyst.

The dynamic inputs in the case of seismic disturbances are ground motion (usually translation), with the horizontal component being more severe than the vertical component. A more detailed discussion of the inputs will follow.

There are two basic computer programs usually required as follows:

1. A static structure program that gives the response of the structure to static loads. We use our 3D structures program (Ref. 5) for this purpose. We use this program to compute our reduced flexibility matrix, which is the required input for the dynamic response program.

2. A dynamic response program that computes the frequencies and mode shapes associated with the reduced flexibility matrix and mass matrix. We use the LUMS program (Ref. 6) to accomplish those computations. The response to time-dependent inputs such as actual earthquake records or frequency-dependent response spectra can then be computed on the basis of an assumed modal damping matrix.

The basic problem in modeling for seismic response is that, although the static structures program can handle structures with a relatively large number of "flexibility" degrees of freedom, the dynamic response program cannot handle systems with an equally large number of "inertial" degrees of freedom. This is so because of the computational difficulties inherent in solving the eigenvalue problem for large systems. For example, our static program (3DS) is capable of handling systems with a large number of beam elements and joints, and the number of flexibility degrees of freedom allowed by 3DS is an order of magnitude higher than the number of inertial degrees of freedom allowed by the dynamics program (LUMS). Although the number of inertial degrees of freedom could be increased, we feel that the "payoff" in terms of increased accuracy of the higher modes with presently available computational methods would be questionable. Our eigenvalue solver is based on a modified Jacobi technique as given in Ref. 7.

MODELING FOR THE NUCLEAR STEAM SUPPLY SYSTEMS

Given a large complex system like the nuclear steam supply system shown in Figs. 1 and 2, we would approach the problem as follows:

We are primarily interested in determining the adequacy of the piping system together with the pipe loads imposed on the nozzles of the components as a result of an earthquake. It would be convenient if we could deal with only half the system, making the assumption that reference plane B (Fig. 1) is a plane of symmetry. If we could make the further assumption that plane A is also a plane of symmetry, then we could represent the mass characteristics of the piping in detail.

In view of the deviations in symmetry, we would examine the dynamic response in two stages:

Stage 1

The first stage would be to represent the complete system by Model A as shown in Figs. 1 and 2. This model has 50 translational degrees of freedom that are summarized in Table 1. Note that we would represent the secondary shield wall by a beam whose cross section may vary such that the stiffness properties are

Table 1 Degrees of Freedom—Model A

Equipment	Typical mass No.	Items per system	Degrees of freedom X	Degrees of freedom Y	Degrees of freedom Z	Total degrees of freedom per system
Reactor	1	1	Yes	Yes	No	2
	2	1	Yes	Yes	No	2
Hot leg	3	2	Yes	Yes	No	4
	4	2	Yes	Yes	No	4
	5	2	Yes	Yes	No	4
Steam generator	6	2	Yes	Yes	No	4
	7	2	Yes	Yes	No	4
	8	2	Yes	Yes	No	4
Cold leg	9	4	Yes	Yes	No	8
Pump	10	4	Yes	Yes	Yes	12
Secondary shield wall	11	1	Yes	Yes	No	2
					Total	50

correct within the limitations of beam theory accounting for shear deformation. The mass of the wall is taken as being concentrated at the same level as the three-link suppressors; the magnitude of the mass should be assigned such that the first mode of the wall is established. Our model assumes that the pumps are connected to the wall through the suppressors by a "superstructure" whose beam-element centroids lie in a horizontal plane. The section properties of the superstructure would follow closely the local wall structure. The computed local distortion of superstructure resulting from the seismic input will give some indication as to the reasonableness of assuming that the wall behaves like a beam.

Note further that Model A picks up only the dynamic response of the system in the horizontal plane, except for the pumps, which include motions in the X, Y, and Z directions.

In Model A the pressurizer is assumed to be decoupled from the remainder of the system. The justification for this assumption is that the size of the pipeline connection from the pressurizer to the hot leg is small compared to the hot-leg pipe size, so that the connection between the pressurizer and the remainder of the system is soft. Accordingly, the response of the pressurizer would be determined separately.

Stage 2

The second stage of the investigation should be aimed at gaining more detailed knowledge of response of the system with particular emphasis on the piping. This would be accomplished by limiting our model for the second stage (Model B) to some portion of the entire system. The precise definition of Model B would depend on the results obtained from Model A. Tentative additional mass points for Model B are indicated in Fig. 2. One approach in modeling for the second stage might be to examine only one-quarter of the system, assuming planes A and B are planes of symmetry, and then to multiply the results by a correction factor determined from stage 1. Note that Model B should include a sufficient number of vertical degrees of freedom so as to allow computation of the response of the system to the vertical component of the earthquake. Seismic stresses should be obtained for earthquakes in the X, Y, and Z directions separately as well as for combined X–Z and Y–Z earthquakes.

MODELING FOR THE PRIMARY LOOP

The primary loop was analyzed by two models shown in Figs. 3 and 4. Model 1 (Fig. 3) allows for 64 inertial degrees of freedom. It should be noted that this model assumes that lateral motion of the steam generator is kinematically constrained and that the pump has no lateral constraints. The model also assumes that the main steam line is anchored to the containment wall and the hot and cold legs are anchored to the reactor which is assumed to move with

the ground. Note that the reactor in this system is supported at the nozzles, whereas the reactor in the nuclear steam supply system shown in Fig. 1 is supported from below by a support skirt. Model 1 of primary loop A assigned three translatory degrees of freedom to each pipe mass joint, two translatory and two rotatory degrees of freedom to the valves, and six degrees of freedom to the steam generator. It should also be noted that equivalent beam properties for the steam generator and coolant-pump closure heads were obtained on the basis of our shell program, Ref. 8, to account for relatively small resistance of the heads to pipe bending loads.

On the basis of the response of Model 1 to frequency-dependent response-spectra inputs (see discussion on "Seismic Inputs"), it became apparent that the primary coolant pump should be laterally braced by snubbers and that dynamic coupling with the building should be considered. More detailed information on the response of the pump impeller and shaft was also required. Accordingly, Model 2 as shown in Fig. 4 has been developed to satisfy these needs. This model has 75 inertial degrees of freedom. The pump impeller shaft and housing are represented by two straight beam elements connected by four bar-linkage arrangements to represent the bearing connections between the shaft and housing. The properties of the four bar linkages have been adjusted so that relative lateral displacement between the shaft and housing is negligible. Relative rotation (beam type) with no moment transmission between the shaft and housing is, however, accounted for by providing universal-type hinge conditions at the ends of the bar-linkage connection to the shaft.

SEISMIC INPUTS

Seismic inputs for a particular nuclear generating site are given by:

1. An "operating-basis* earthquake" intended to represent the maximum probable earthquake that is expected to occur.

2. A "design-basis* earthquake" that represents the upper limit of potential hazard.

*The "operating-basis earthquake" has been up until recently referred to as the "design earthquake," and the "design-basis earthquake" has been referred to as a "maximum-potential earthquake." This terminology does not correspond completely with that being developed for inclusion in the AEC regulatory criteria.

There are several ways in which the earthquake may be defined:

1. The seismic input can be defined as frequency-dependent response spectra that are a plot of the maximum values of some parameter, such as acceleration, experienced by a hypothetical single-degree-of-freedom oscillator with a given fraction of critical damping that is subject to an actual earthquake time-history record (accelerogram) as a function of the oscillator natural period of vibration.

A single point on a response-spectra curve for a given fraction of critical damping, η, is obtained by determining the time-history response of an oscillator with damping η and a given natural frequency to base motion defined by the accelerogram. The time-history response can be obtained by an analog computer. The maximum response of this oscillator then determines the ordinate of the response spectra, and the abscissa is the natural period of vibration of the oscillator. The process is then repeated for other assumed periods. The resulting response-spectra curve is quite jagged, and the process may be repeated for several earthquake records to obtain an average. The response-spectra curves presented to the analyst are based on a smooth envelope of such curves.

2. Another approach is to specify a time-dependent input in the form of a specific accelerogram such as the 1950 El Centro record or a "standard" acceleration time history based on artificial earthquake motions simulated by computers.

3. Finally, the seismic input can be defined in terms of a random wave function based upon a large number of artificially generated accelerograms. The input is expressed in terms of a frequency-dependent power spectral-density function that expresses the relative values of energy in each frequency component of the earthquake.

Of the three kinds of inputs, the first (response-spectra curves) is easiest to use because computer input can be easily prepared. The computed response of the structure is in the form of acceleration values in each inertial degree of freedom for each natural frequency of the system. The usual practice is to take the root mean square (rms) of response of the system to the significant modal contributions. There is no assurance that this procedure is conservative. On the other hand, to take the absolute sum of the responses would be overly conservative. In addition, the dynamic character of the input is not retained in the response. These appear to be disadvantages.

The second method (time-history inputs) requires considerably more in the way of computer input preparation and computer time. The advantage here is that response-spectra curves for secondary equipment mounted at specified locations can be constructed from the time-history response.

The third method (random wave inputs) seems most attractive because the probabilistic nature of the input lends itself to reliability studies. The input-preparation and computer time should not be as high as in the second method. Another advantage of this method is that the response of the system does not lose its dynamic character as does occur in the first method. This allows one to compute the power spectral-density function associated with a particular inertial degree of freedom of the structure which can then be used as dynamic input to equipment items such as small pumps and motors or the internals of massive components such as the steam generator or coolant pump.

SUMMARY OF SEISMIC-ANALYSIS PROCEDURE

A step-by-step procedure for determining the seismic response to response-spectra inputs may be summarized as follows:

1. The system is represented by a lumped-mass model that can have translational as well as rotational degrees of freedom.

2. A static analysis of the structural system is utilized to obtain a reduced flexibility matrix corresponding to the degrees of freedom established in step 1. The structural system can be represented by an assemblage of beam elements. The natural frequencies and mode shapes of the system are then obtained. The eigenvalue problem may be solved by a modified Jacobi technique, Ref. 9. Participation factors and modal effective masses should also be computed.

3. The peak displacements and accelerations of each degree of freedom due to each significant modal response of the system are determined. The magnitude of the modal response is determined from the response-spectra curve corresponding to a particular ratio of critical modal damping.

4. The response of the system to the significant modal contributions is combined by taking the root mean square (rms) of the values determined in step 3.

5. The reversed effective forces (or moments) corresponding to the peak rms accelerations determined in step 4 are applied to the system statically, and the resulting displacements, joint loads, and stresses are determined throughout the structure by the static structures program.

In connection with the above, it is important to note:

1. Seismic inputs in the form of time histories can also be handled, as will be explained later.

2. Rocking motion of the containment building may occur when soil–structure interaction is considered; this can also be handled, as will be subsequently explained.

3. The computation of the reduced flexibility matrix which is used in determining the dynamic response should not impose unrealistic constraints on the system. For example, if we wish to compute the dynamic response of a cantilever beam by lumped-mass model and to neglect the rotational mass effects in computing the dynamic response, the rotational flexibility, which is coupled with translatory motion, should, nevertheless, be accounted for in the model.

4. The static structures program should account for the additional contribution in displacements brought about by shearing strain. For example, the motion of a shielding wall is probably dominated by the shear contribution. The program should also account for the increased flexibility in the piping elbows (see Ref. 10) due to flattening of the cross section.

5. A program based on shell theory should be used to compute equivalent beam properties for shell closure heads connecting the nozzles to the vessels.

THEORETICAL BASIS FOR COMPUTING THE RESPONSE OF A LUMPED-MASS SYSTEM TO SEISMIC INPUTS

Let $\tilde{q}$ be a column vector whose elements represent the generalized coordinates of a system relative to a reference frame R which is fixed to the base of a structure whose stiffness matrix is given by $\tilde{K}$. Let $\tilde{M}$ be a mass matrix whose elements correspond to the elements of $\tilde{q}$. Assume that m of the degrees of freedom in $\tilde{q}$ have associated mass values equal to zero and let those degrees of freedom be represented by q_0 such that:

$$(\tilde{q})_N = \begin{pmatrix} q \\ q_0 \end{pmatrix}_{N=n+m}$$

and

$$\tilde{M} = \begin{pmatrix} M & 0 \\ 0 & 0 \end{pmatrix}_N \qquad M = \begin{pmatrix} m_{11} & \dots & m_{1n} \\ \vdots & & \vdots \\ m_{n1} & \dots & m_{nn} \end{pmatrix}_n$$

If static forces, $\tilde{F}$, are applied to the structure, then these forces will satisfy the equation:

$$\tilde{F} = \tilde{K}\tilde{q} = \begin{pmatrix} K_{11} & K_{12} \\ K_{21} & K_{22} \end{pmatrix} \begin{pmatrix} q \\ q_0 \end{pmatrix} \quad (1)$$

or

$$\begin{aligned} (F)_n &= K_{11}q + K_{12}q_0 \\ (F_0)_m &= K_{21}q + K_{22}q_0 \end{aligned} \quad (2)$$

where

$$\tilde{F} = \begin{pmatrix} F \\ F_0 \end{pmatrix}$$

If the system undergoes an absolute acceleration represented by the column vector

$$\tilde{a} = \begin{pmatrix} \ddot{\xi} \\ a_0 \end{pmatrix}$$

where a_0 are accelerations associated with the massless coordinates, then by D'Alembert's principle the system can be considered to be in equilibrium under applied reversed effective forces:

$$\begin{aligned} (F)_n &= -M\ddot{\xi} \\ (F_0)_m &= \begin{pmatrix} \diagdown 0 \diagdown \end{pmatrix} (a_0) = 0 \end{aligned} \quad (3)$$

Substituting Eq. 3 into Eq. 2:

$$\begin{aligned} -M\ddot{\xi} &= K_{11}q + K_{12}q_0 \\ 0 &= K_{21}q + K_{22}q_0 \end{aligned} \quad (4)$$

so that

$$-M\ddot{\xi} = (K_{11} - K_{12}\,K_{22}^{-1}\,K_{21})\,q = F$$

If we define:

$$K = (K_{11} - K_{12}\,K_{22}^{-1}\,K_{21}) \quad (5)$$

then

$$F = Kq = -M\ddot{\xi} \quad (6)$$

The relative displacements can also be expressed in terms of the flexibility matrix as follows:

$$\begin{pmatrix} q \\ q_0 \end{pmatrix} = \begin{pmatrix} C_{11} & C_{12} \\ C_{21} & C_{22} \end{pmatrix} \begin{pmatrix} F \\ F_0 \end{pmatrix} \quad (7)$$

The term $F_0 = 0$ if the system is subject only to inertia forces, in which case:

$$q = C_{11}F \quad (8)$$

$$q_0 = C_{21}F \quad (9)$$

If we define:

$$C_{11} = C \quad (10)$$

then a comparison between Eqs. 6 and 8 will reveal that

$$C = K^{-1} \quad (11)$$

where C represents the reduced flexibility matrix. We generate C by applying a unit load corresponding to each inertial degree of freedom $q_1, \ldots, q_n$, and computing the resulting displacement vector q by our static structures program. Note that it is easier to generate the "dynamic" flexibility matrix C than it would be to obtain the dynamic stiffness matrix K, since, as can be seen from Eq. 5, K involves the inverse of K_{22} which cannot be economically computed directly for large values of m.

BASE MOTION

The absolute acceleration column vector $\ddot{\xi}$ can be expressed as:

$$\ddot{\xi} = \ddot{q} + \ddot{\zeta} \quad (12)$$

where $\ddot{q}$ represents the acceleration of the inertial degrees of freedom relative to the base and $\ddot{\zeta}$ represents the additional acceleration due to the base motion.

Substituting Eq. 12 into Eq. 6, we obtain:

$$M\ddot{q} + Kq = -M\ddot{\zeta} \quad (13)$$

From Eq. 13 it is apparent that the problem can be viewed as a stationary base structure which is subject to time-dependent forces $-M\ddot{\zeta}$.

The discussion following is aimed at determining the elements of $\ddot{\xi}$ for different types of ground motion. The foundation is assumed to be a rigid base on which the structure (primary piping system and shielding or containment if required) is mounted. It is assumed that the motion of the base will be prescribed in one of the three forms of seismic inputs described previously.

Consider the system shown in Fig. 5. The base of the structure is assumed fixed to reference frame R, which in turn moves with respect to the Newtonian reference frame Z. Points on the structure are referred to the global coordinates x_1, x_2, and x_3, which are "embedded" in the base. The base of the structure is assumed to translate an amount $\mathbf{r}$ and also to undergo a small rotation $\boldsymbol{\theta}$. The components of $\mathbf{r}$ and $\boldsymbol{\theta}$ can be referred to R by Z_1, Z_2, and Z_3 and θ_1, θ_2, and θ_3, respectively.

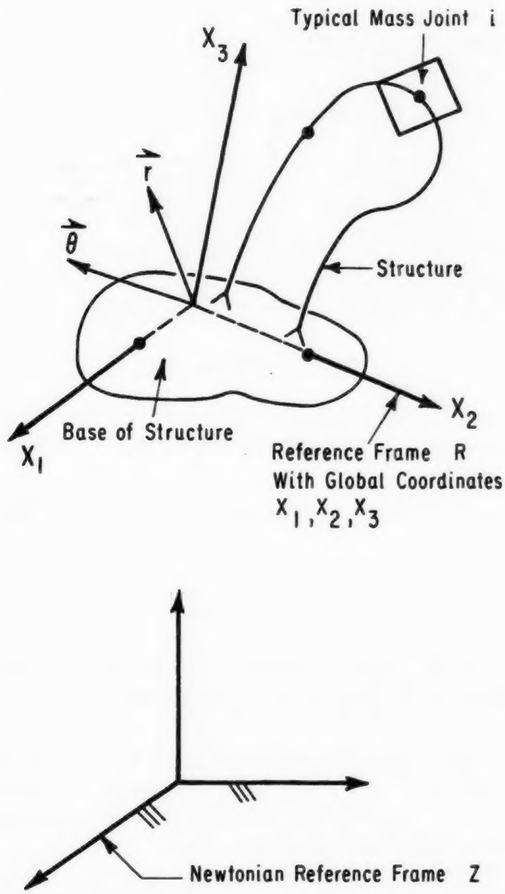


Fig. 5 Reference frames.

Consider a typical mass joint i as shown in Fig. 5. It has associated with it a mass matrix:

$$(M_{jk}^{(i)})$$

which is diagonal only if the principal inertia axes of mass i coincide with the global coordinate directions x_1, x_2, and x_3; otherwise off-diagonal terms appear.

Let the relative displacements of the ith mass be:

$$u_j^{(i)}$$

where j can take on values from 1 to 6 depending on the inertial degrees of freedom assigned to the ith mass joint.

Similarly the absolute displacement of the ith mass will be:

$$\psi_j^{(i)}$$

Note the correspondence:

$$q_s \longleftrightarrow u_j^{(i)}$$

$$\xi_s \longleftrightarrow \psi_j^{(i)}$$

As can be seen from Eq. 13, so long as the system remains elastic (i.e., K = constant), the equations governing the dynamic response are linear, so that it is permissible to find the response of the system by superposing the effects of each component of the base motion.

Let the base translate an amount $z_k(t)$. Then the ith mass joint will accelerate the absolute amount represented by components:

$$\ddot{\psi}_j^{(i)} = \ddot{u}_j^{(i)} + \delta_{kj}\, \ddot{z}_k(t) \tag{14}$$

where

$$\delta_{kj} = \begin{cases} 1 & \text{if } j = k \\ 0 & \text{if } j \neq k \end{cases}$$

so that for translatory motion of the base

$$\ddot{\xi}_s = \delta_k^s\, \ddot{z}_k(t) \tag{15}$$

where

$$\delta_k^s = \begin{cases} 1 & \text{if } q_s \text{ corresponds to a displacement in the } k\text{th direction} \\ 0 & \text{otherwise} \end{cases} \tag{15a}$$

If the base rotates a small amount as, for example, an amount θ_1, then the resulting displacements of the ith mass joint due to the rocking motion are shown in Fig. 6. By assuming that the base rotation is small, we find that for rocking motion:

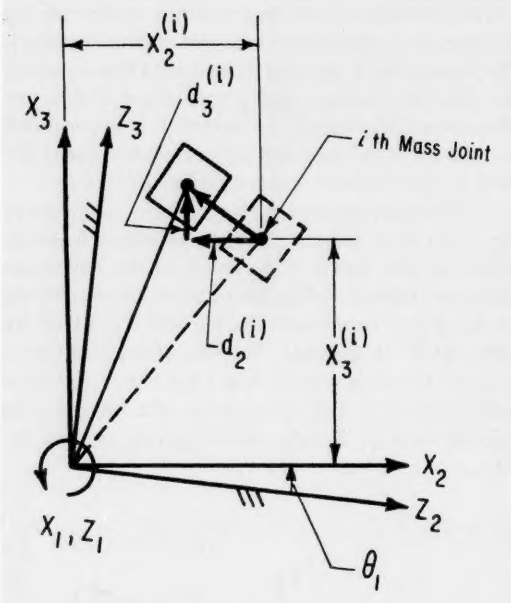
$$\ddot{\zeta}_s = \left(\Delta_k^s - \sum_{j=1}^{3} \epsilon_{ijk}^s x_j\right) \ddot{\theta}_k(t) \qquad \text{(no sum on } k\text{)} \tag{16}$$

where

$$\Delta_k^s = \begin{cases} 1 & \text{if } q_s \text{ represents a rotation in the } k\text{th direction} \\ 0 & \text{otherwise} \end{cases}$$

$$\epsilon_{ijk}^s = \begin{cases} \left.\begin{array}{ll} 1 & \text{if } ijk \text{ are unequal and in sequential order} \\ -1 & \text{if } ijk \text{ are unequal and not in sequential order} \\ 0 & \text{if any two indices are equal} \end{array}\right\} & \text{and if } q_s \text{ represents a displacement in the } i\text{th direction} \\ 0 & \text{otherwise} \end{cases}$$

x_j = the global coordinates of mass joint corresponding to q_s
k = the axis about which the base rotates



$$\left.\begin{array}{l} d_2^{(i)} = -\,x_3^{(i)}\,\theta_1 \\ d_3^{(i)} = x_2^{(i)}\,\theta_1 \\ d_4^{(i)} = \theta_1 \end{array}\right\}$$

Absolute displacements of the ith mass due to base motion θ_1 and asuming the structure is rigid.

Fig. 6 Base motion = θ_1.

DETERMINATION OF FREQUENCIES AND MODE SHAPES

The natural frequencies and mode shapes of the system are determined from Eq. 13 with the right-hand side set equal. Note that this implies damping in the system is sufficiently small (less than 20% of critical) so as not to influence the results. The theory and computational procedure will not be discussed in detail because the problem is adequately covered in the literature. As mentioned previously, we determine the frequencies and mode shapes by use of a modified Jacobi technique which requires that the operator matrix be symmetric. In our problem the homogeneous equation is

$$M\ddot{q} + Kq = 0 \tag{13a}$$

or

$$CM\ddot{q} + q = 0 \tag{17}$$

If M contains off-diagonal terms, then we can always find a triangular matrix B such that

$$B^T B = M \tag{18}$$

(B^T is the transpose of B)

and make the transformation

$$y = Bq \rightarrow q = B^{-1} y \tag{19}$$

Substituting Eq. 19 into Eq. 18, we obtain

$$BCB^T\ddot{y} + y = 0 \qquad (20)$$

It can be shown that BCB^T is a symmetric matrix.

The homogeneous solutions of Eq. 20 can be found if we let

$$y = \psi e^{i\Omega t} \qquad (21)$$

Substituting into Eq. 23, we arrive at

$$(BCB^T)\psi = \left(\frac{1}{\Omega^2}\right)\psi \qquad (22)$$

which can now be solved for the eigenvalues $1/\Omega^2$ and eigenvectors ψ by operating on the symmetric matrix BCB^T.

The relation between the mode shapes ϕ, natural frequencies ω of Eq. 13*a*, and ψ,Ω can be found by noting that

$$q = \phi e^{i\omega t} = B^{-1}y = B^{-1}\psi e^{i\Omega t}$$

Hence

$$\left.\begin{aligned}\psi &= B\phi\\ \Omega &= \omega\end{aligned}\right\} \qquad (23)$$

TRANSFORMATION TO PRINCIPAL COORDINATE SYSTEM

A set of decoupled equations can be derived from Eq. 13 by the transformation:

$$q = \Phi p \qquad (24)$$

where

$$\Phi = (\phi_1, \phi_2, \ldots, \phi_n)$$

is an nxn matrix whose columns are the mode shapes associated with the frequencies ω_i, $i = 1, \ldots, n$ which gives rise to the set of decoupled equations:

$$\ddot{p}_j + \omega_j^2 p_j = -\sum_{k,l=1}^{m} \phi_{kj} m_{kl} \ddot{\xi}_l \qquad (25)$$

provided that the mode shapes have been orthonormalized with respect to the mass matrix such that:

$$\Phi^T M \Phi = \begin{pmatrix} 1 & & 0 \\ & \ddots & \\ 0 & & 1 \end{pmatrix}$$

COMPUTATION OF THE SYSTEM RESPONSE

Equation 25 serves as a basis for computing the response of the system. It is at this point that the assumption of viscous damping is introduced to the problem. There are only two forms of physically identifiable damping that will yield a decoupled set of equations after the transformation $q = \Phi p$, namely, uniform mass damping $C_m = 2\beta M$ and uniform structural damping $C_s = \alpha K$.

It is our practice to use a modal damping definition which is a combination of mass and structural damping and which, by analogy to a one-degree-of-freedom oscillator, is expressed as a fraction of critical damping η_j in each mode.* The dynamic equations are then expressed in the principal coordinate system by

$$\ddot{p}_j + 2\eta_j\omega_j\dot{p}_j + \omega_j^2 p_j = -\sum_{k,l=1}^{m} \phi_{kj} m_{kl} \ddot{\xi}_l \qquad (26)$$

For translatory motion of the base in the ith direction,

$$\ddot{\xi}_l = \delta_i^l \ddot{z}_i(t) \qquad (15)$$

If we assume that $\ddot{z}_i(t) = P_i\ddot{z}(t)$, where P_i is the maximum value of $\ddot{z}_i(t)$, then we have

$$\ddot{p}_j + 2\eta_j\omega_j\dot{p}_j + \omega_j^2 p_j = -P_i\left(\sum_{k,l=1}^{m} \phi_{kj} m_{kl} \delta_i^l\right)\ddot{z}(t) \qquad (26a)$$

In Eq. 26*a*, the expression

$$\Gamma_j = \sum_{k,l=1}^{m} \phi_{kj} m_{kl} \delta_i^l \qquad (27)$$

is known as the participation factor for the jth mode. The ratio $\Gamma_j/\Sigma\Gamma_j$ represents that portion of the total

*See Ref. 11 for typical values of damping. If some portion of the structure has different damping characteristics, then the modal damping can be "weighted" on the basis of absolute value of the mode shapes.

earthquake input which excites the jth mode of the system. The solution to Eq. 26a for zero initial conditions ($p_j = \dot{p}_j = 0$) is given by*

$$p_i(t) = \frac{-P_i\Gamma_j}{\omega_j^2} D_j(t) \tag{28}$$

where

$$D_j(t) = \omega_j \int_0^t e^{-\eta_j\omega_j(t-\tau)} \sin[\omega_j(t-\tau)]\,\ddot{z}(\tau)\,d\tau \tag{29}$$

is known as the dynamic load factor and η_j^2 is negligible compared to 1.

Note that, if the seismic input in the ith direction is in the form of a time history, $P_i\ddot{Z}(t)$, then the response of the system can be computed by numerical integration of Eq. 29.

Successive differentiations of Eqs. 28 and 29 give

$$\dot{p}_j(t) = \frac{1}{\omega_j}[P_i\Gamma_j\eta_j D_j(t) - P_i\Gamma_j D_j^*(t)] \tag{30}$$

$$\ddot{p}_j(t) = P_i\Gamma_j D_j(t) - P_i\Gamma_j\ddot{Z}(t) + P_i\Gamma_j 2\eta_j D_j^*(t)$$

$$= \Gamma_j[P_2\, D_j(t) + 2\eta_j P_i D_j^*(t)] - \Gamma_j P_i \ddot{Z}(t) \tag{31}$$

where

$$D^*(t) = \omega_j \int_0^t e^{-\eta_j\omega_j(t-\tau)} \cos[\omega_j(t-\tau)]\,\ddot{z}(\tau)\,d\tau \tag{32}$$

It can be shown that

$$\ddot{\xi}_k^j = \varphi_{kj}\Gamma_j f_j(t)$$

represents the absolute acceleration of the kth degree of freedom in the jth mode, where

$$f_j(t) = P_i D_j(t) + 2\eta_j P_i D_j^*(t)$$

At some particular time, t_1, $|f_j(t)|$ takes on a maximum value which is designated

$$S_a^{(j)}$$

so that

$$(\ddot{\xi}_k^j)_{\max.} = \varphi_{kj}\Gamma_j S_a^{(j)} \tag{33}$$

$S_a^{(j)}$ represents the response-spectra acceleration value corresponding to a frequency of ω_j and a fraction of critical damping η_j which can be picked off the response-spectra curves supplied to the analyst.

If all $f_j(t)$ took on a maximum value at the same time, the maximum absolute acceleration in the kth degree of freedom would be:

$$\left|\ddot{\xi}_k\right|_{\max.} = \sum_j \left|\varphi_{kj}\Gamma_j S_a^{(j)}\right|$$

which would represent an upper bound and would lead to extremely conservative results. In line with what others have done, we designate the acceleration by:

$$\left|\ddot{\xi}_k\right|_{\text{rms}} = \left[\sum_j \left|\varphi_{kj}\Gamma_j S_a^{(j)}\right|^2\right]^{1/2} \tag{34}$$

There is, however, no assurance that this method is conservative.

MODAL EFFECTIVE MASS

If the structure base is subject to an earthquake in the ith direction, then a shear force will be developed at the base of the structure that will be equal to the sum of the reversed effective forces in the ith direction. This sum for a single-degree-of-freedom system can be expressed by

$$H = -ma(t) \tag{35}$$

where $a(t)$ is the acceleration of the mass.

For our system the force in the kth degree of freedom is given by

$$F_k = -\sum_l m_{kl}\ddot{\xi}_l$$

The sum of reversed effective forces in the ith direction (i.e., the shear force at the base) would then be

$$H = \sum_k F_k\delta_i^k = -\sum_{k,l} m_{kl}\delta_i^k\,\ddot{\xi}_l$$

(Text continues on page 316.)

*The solution is based on a Laplace-transform technique involving the convolution integral as given in Sec. 8.8 of Ref. 12.

Table 2 Piping Earthquake Analysis: Details of Pump and Building (75 Degrees of Freedom)

MODE NO= 2 NAT.FREQUENCY= .63582+02 RAD/SEC PERIOD= .98821-01SEC

DEG.OF FREEDOM	JOINT NO.	DIRECTION	NORMALIZED MODE SHAPE
1	4	1	.100024-05
2	4	2	-.113269-05
3	4	3	-.210947-03
4	6	1	.211412-05
5	6	2	-.252993-05
6	6	3	-.445119-03
7	9	1	.125799-03
8	9	2	-.669758-04
9	9	3	-.445565-03
10	9	4	.189131-05
11	9	5	.362958-05
12	62	1	.249901-03
13	62	2	-.212063-04
14	62	3	.279784-04
15	62	4	.572767-06
16	62	5	.661314-05
17	16	1	.229951-03
18	16	2	-.194786-04
19	16	3	.213627-04
20	16	4	.572786-06
21	16	5	.661332-05
22	69	1	.529012-04
23	69	2	-.250348-04
24	69	3	.207060-04
25	20	1	-.705380-03
26	20	2	-.977861-05
27	20	3	.763737-04
28	73	1	.766976-03
29	73	2	-.316085-04
30	73	3	.206619-04
31	21	1	-.790361-03
32	21	2	-.338259-04
33	21	3	.118842-03
34	68	1	.134784-02
35	68	2	-.198880-04
36	68	3	.215542-04
37	68	4	-.348561-06
38	68	5	.970103-05
39	77	1	.163847-02
40	77	2	-.128564-04
41	77	3	.206271-04
42	23	1	-.157974-03
43	23	2	.201908-04
44	23	3	.163414-03
45	83	1	.223351-02
46	83	2	.121715-04
47	83	3	.206291-04
48	85	1	.398147-03
49	85	2	.344208-06
50	86	1	.140976-02

Table 3 Run No. 2: Frequencies and Acceleration Spectra (0.5% Damping)

Mode	Natural frequency rads/sec	Natural frequency cycles/sec	Period, sec	Acceleration* spectrum Gal†	Acceleration* spectrum in./sec^2
1	24.933	3.968	0.252	1455	562.212
2	45.874	7.301	0.137	1560	602.784
3	48.641	7.741	0.129	1560	602.784
4	71.617	11.398	0.0877	1525	589.26
5	76.269	12.139	0.0824	1505	581.532
6	84.178	13.397	0.0746	1420	548.688
7	95.523	15.202	0.0658	1280	494.592
8	101.432	16.143	0.0619	1160	448.224
9	105.364	16.769	0.0596	1000	386.4
10	121.309	19.307	0.0518	680	262.752

*These values correspond to $S_a^{(j)}$ given in Eq. 33.
†One gal = 0.3864 in./sec^2.

Table 4 Piping Earthquake Analysis: Response for Excitation Along X1 (Combined Effect of 10 Modes)

Degree of freedom	Inertia force/moment	Degree of freedom	Inertia force/moment
1	0.90625603+03	2	0.14188012+04
3	0.17628831+04	4	0.21580414+04
5	0.34611893+04	6	0.36069968+04
7	0.85133734+04	8	0.12455804+05
9	0.10595503+05	10	0.14027295+06
11	0.11412890+06	12	0.74729408+04
13	0.13877705+06	14	0.10275893+06
15	0.54254971+04	16	0.10966526+06
17	0.13061236+06	18	0.24177501+05
19	0.35779762+05	20	0.29734077+04
21	0.18162747+05	22	0.21230245+05
23	0.64832845+04	24	0.52319521+04
25	0.11769135+04	26	0.10191121+05
27	0.63073424+04	28	0.28718860+04
29	0.47622944+04	30	0.13086668+04
31	0.50322855+03	32	0.15129174+03
33	0.15758350+03	34	0.34778471+02
35	0.11150976+04	36	0.17693438+04
37	0.19690824+04	38	0.36513036+04
39	0.52395476+04	40	0.58590699+04
41	0.12771626+05	42	0.94269668+04
43	0.35077760+04	44	0.14569370+06
45	0.68465313+05	46	0.28453231+05
47	0.15756949+08	48	0.33448137+08
49	0.50588401+07	50	0.52533089+04
51	0.79810253+04	52	0.66713167+03
53	0.84427454+03	54	0.17705107+03
55	0.82730635+03	56	0.16426039+05
57	0.57776180+04	58	0.18665582+04
59	0.22267845+04	60	0.21720549+04
61	0.74838679+03	62	0.14519177+03
63	0.28171509+03	64	0.72700775+02

Table 5 Piping Earthquake Analysis: Details of Pump and Building (Disturbance in Y Direction)

RESULTING DISPLACEMENTS FOR LOAD NUMBER 101

JNT NO.	DISPLACEMENT 1	DISPLACEMENT 2	DISPLACEMENT 3	ROTATION 1	ROTATION 2	ROTATION 3	NO.
1	.00000000	.00000000	.00000000	.00000000	.00000000	.00000000	1
2	.49744161-05	.35862989-05	-.15703616-03	.71214537-05	-.61068824-05	-.27199448-07	2
3	.23746662-04	.19573141-04	-.15590516-02	.40626637-04	-.28470799-04	-.11436849-06	3
4	.41759821-04	.36874842-04	-.39563678-02	.64032184-04	-.33340563-04	-.10658108-06	4
5	.60785187-04	.51833322-04	-.66065468-02	.76487008-04	-.19241954-04	-.27546139-07	5
6	.81776684-04	.63386155-04	-.85764118-02	.77140017-04	.15299247-04	.99027520-07	6
7	.94919804-04	.66981634-04	-.90965082-02	.71780948-04	.44687188-04	.17503311-06	7
8	.98713775-04	.67749780-04	-.91236392-02	.69433013-04	.54350862-04	.19312570-06	8
9	.35108881-02	-.18276203-02	-.83929498-02	.56411246-04	.99683548-04	.24493241-06	9
10	.11351654-03	.68044448-04	-.84061520-02	.56073482-04	.98545894-04	.24493241-06	10
11	.12042803-03	.58167115-04	-.62102207-02	.32789624-04	.14998601-03	.14642178-06	11
12	.12262825-03	.55204704-04	-.38620517-02	.20683814-04	.17510350-03	.21214429-07	12
62	.68546952-02	-.64218273-03	.28177447-02	.18524556-04	.17917473-03	.35513578-12	13
16	.63101955-02	-.58885688-03	.26379216-02	.18519348-04	.17918667-03	.30425226-12	14
63	.73853270-02	-.69996712-03	.28171093-02	.18518996-04	.17918729-03	.35513578-12	15
69	.23819730-02	-.11812645-02	.26313239-02	-.87119076-05	.19701137-03	.00000000	16
17	.12824264-03	.50056042-04	.26379152-02	.18521546-04	.17918156-03	.00000000	17
64	.73853270-02	-.69996713-03	.26379220-02	.18518996-04	.17918729-03	.35716569-12	18
18	-.59370907-02	.12032982-02	.26733298-02	.24431161-04	.17588822-03	-.31519486-05	19
65	.88188411-02	-.84811099-03	.26379225-02	.18518549-04	.17918807-03	.46993904-12	20
70	.88188411-02	-.84811099-03	.26280905-02	-.14436771-04	.21367371-03	.00000000	21
19	-.73802791-02	.14642605-02	.26768738-02	.29022186-04	.17348862-03	-.64607731-05	22
71	.88188411-02	-.84811099-03	.26218558-02	.18518547-04	.17918807-03	.16682347-11	23
72	.88188411-02	-.84811099-03	.26397744-02	.18518548-04	.17918807-03	.12688103-11	24
66	.12313055-01	-.12091978-02	.26379237-02	.18517563-04	.17918974-03	.72457287-12	25
20	-.11792352-01	.69648397-02	.98855557-02	.10504437-03	.11979681-03	-.56030926-04	26
73	.21544494-01	.45670049-03	.26569875-02	-.30751920-04	.25832091-03	.00000000	27
67	.25506724-01	.56780310-03	.27022803-02	-.55024988-04	.29723296-03	.25880628-07	28
21	-.79895877-02	.81507249-02	.20223216-01	.71222555-04	.11017021-03	-.24216196-04	29
74	.35969398-01	.25047272-02	.26760741-02	-.64161971-04	.35883701-03	.00000000	30
15	.35969398-01	.25047246-02	.27022824-02	-.55025747-04	.29723396-03	.25881088-07	31
75	.35969395-01	.25047272-02	.26670568-02	-.55025747-04	.29723396-03	.25886478-07	32
76	.35969395-01	.25047246-02	.26967798-02	-.55025747-04	.29723396-03	.25884680-07	33
68	.39684839-01	.31925562-02	.27022831-02	-.55025903-04	.29723411-03	.25881193-07	34
22	-.52064523-02	.90388027-02	.22099221-01	-.46984071-04	.90680536-04	.48155645-04	35
77	.49449463-01	.50261143-02	.26849627-02	-.59926656-04	.31881458-03	.00000000	36
78	.63909420-01	.76772212-02	.26832778-02	-.50136188-04	.28402854-03	.00000000	37
13	.63909420-01	.76772187-02	.27022840-02	-.55026346-04	.29723434-03	.25881876-07	38
82	.63909420-01	.76772200-02	.26907715-02	-.55026346-04	.17666854-03	.25947319-07	39
14	.57512920-01	.64930490-02	.27022838-02	-.55026330-04	.29723426-03	.25881696-07	40
23	-.24303005-02	.10682583-01	.22102357-01	-.45510308-04	.10661659-03	.57366116-04	41
79	.63909419-01	.76772212-02	.26731585-02	-.17558831-03	.29723434-03	.26042350-07	42
83	.66712794-01	.81702311-02	.26880137-02	-.57889590-04	.32913287-03	.00000000	43
81	.63909417-01	.76772187-02	.26967814-02	-.55026346-04	.29723434-03	.25888328-07	44
80	.63909417-01	.76772212-02	.26670584-02	-.55026348-04	.29723434-03	.25891554-07	45
87	.57497476-01	.65294338-02	.93581571-07	-.56081995-05	.55772797-04	-.10946025-05	46
24	.14282203-02	.11852339-01	.21231352-01	-.12033219-05	.21719696-03	.33003002-04	47
88	.57513928-01	.64934771-02	-.19162838-02	-.56081995-05	.55772797-04	-.10946035-05	48
89	.57498483-01	.65298617-02	.16741136-04	-.56081995-05	.55772797-04	-.10946025-05	49

Table 6 Piping Earthquake Analysis: Details of Pump and Building (Disturbance in *X* Direction)

SEG. NO.	JOINT NO.	FORCES IN LBS. F(X)	F(Y)	F(Z)	MOMENTS IN FOOT-LBS. M(X)	M(Y)	M(Z)	CODE COMBINED STRESS (PSI)	TYPE
1	1	-3161.	-1874.	8983.	-59250.	63738.	282.	104.	TANGNT
	2	3161.	1874.	-8983.	49599.	-47017.	-191.	528.	
2	2	-3161.	-1874.	8983.	-49599.	47017.	191.	528.	TANGNT
	3	3161.	1874.	-8983.	35283.	-22219.	-56.	322.	
3	3	-3161.	-1874.	8983.	-35283.	22219.	56.	322.	TANGNT
	4	3161.	1874.	-8983.	20967.	2579.	78.	163.	
4	4	-3133.	-1833.	10497.	-20967.	-2579.	-78.	163.	TANGNT
	5	3133.	1833.	-10497.	4238.	31557.	146.	246.	
5	5	-3133.	-1833.	10497.	-4238.	-31557.	-146.	246.	TANGNT
	6	3133.	1833.	-10497.	-12492.	60534.	213.	478.	
6	6	-3081.	-1750.	13448.	12492.	-60534.	-213.	478.	TANGNT
	7	3081.	1750.	-13448.	-24572.	81457.	169.	658.	
7	7	-3081.	-1750.	13448.	24572.	-81457.	-169.	658.	TANGNT
	8	3081.	1750.	-13448.	-27934.	87284.	158.	709.	
8	8	-3081.	-1750.	13448.	27934.	-87284.	-158.	450.	TANGNT
	10	3081.	1750.	-13448.	-45024.	116887.	96.	852.	
9	9	2691.	1853.	8437.	5982.	9482.	0.	19.	TANGNT
	10	-2691.	-1853.	-8437.	-732.	-17106.	0.	30.	
10	10	-390.	103.	21885.	45756.	-99780.	-96.	746.	TANGNT
	11	390.	-103.	-21885.	-73568.	147954.	-625.	811.	
11	11	-390.	103.	21885.	73568.	-147954.	625.	1162.	TANGNT
	12	390.	-103.	-21885.	-91236.	178556.	-1083.	450.	
12	12	-390.	103.	21885.	91236.	-178556.	1083.	293.	TANGNT
	17	390.	-103.	-21885.	-79240.	262611.	-1263.	400.	
13	17	-7098.	-10822.	-5225.	-93990.	53961.	31976.	292.	TANGNT
	18	7098.	10822.	5225.	61884.	-32905.	-31976.	199.	
14	18	-7098.	-10822.	-5225.	-61884.	32905.	31976.	238.	TANGNT
	19	7098.	10822.	5225.	54308.	-27937.	-31976.	416.	
15	19	-7098.	-10822.	-5225.	-54308.	27937.	31976.	604.	BEND
	20	7098.	10822.	5225.	-13065.	-3757.	9463.	153.	
16	20	-1861.	-9841.	-4022.	13065.	3757.	-9463.	109.	TANGNT
	21	1861.	9841.	4022.	-40396.	-11398.	40811.	387.	

Table 7 Details of Bar 41

RESULTANT FORCE VECTOR ON BAR ENDS (BAR COORDINATES)

BAR END	FORCE 1 LB.	FORCE 2 LB.	FORCE 3 LB.	MOMENT 1 IN-LB.	MOMENT 2 IN-LB.	MOMENT 3 IN-LB.
1	-.14133480+05	.26464660+05	.14122323+05	-.24306198+07	-.61971091+06	-.14499003+07
2	.14133480+05	-.26464660+05	-.14122323+05	.19265942+07	.61971091+06	.94547651+06

RESULTANT FORCE VECTOR ON BAR ENDS (SYSTEM COORDINATES)

BAR END	FORCE 1 LB.	FORCE 2 LB.	FORCE 3 LB.	MOMENT 1 IN-LB.	MOMENT 2 IN-LB.	MOMENT 3 IN-LB.
1	-.14122323+05	-.14133480+05	-.26464660+05	.14499003+07	-.24306198+07	.61971091+06
2	.14122323+05	.14133480+05	.26464660+05	-.94547651+06	.19265942+07	-.61971091+06

RESULTANT DISPLACEMENT VECTOR ON BAR ENDS

BAR END	DISPLACEMENT 1 INCH	DISPLACEMENT 2 INCH	DISPLACEMENT 3 INCH	ROTATION 1 RADIAN	ROTATION 2 RADIAN	ROTATION 3 RADIAN
1	.47973946-02	-.19218484-01	-.11533037+00	.24413275-03	-.34552202-04	-.23003661-04
2	.12161493-01	-.22110827-01	-.13233921-00	.25869349-03	-.29167807-04	-.14998902-04

STRESSES, SHEAR FORCES, AND TWISTING MOMENT AT INTERMEDIATE BAR SECTIONS

DISTANCE FROM END 1 INCH	NORMAL STRESSES AT CROSS SECTION CORNERS CORNER 1 PSI	CORNER 2 PSI	CORNER 3 PSI	CORNER 4 PSI	SHEAR FORCE AXIS 1 LB.	SHEAR FORCE AXIS 3 LB.	TWISTING MOM. AXIS 2 IN-LB.
.000000	830.36	4247.32	3383.75	-33.20	-14133.48	14122.32	-619710.91
1.784500	852.57	4225.13	3361.55	-11.01	-14133.48	14122.32	-619710.91
3.568999	874.78	4202.94	3339.34	11.18	-14133.48	14122.32	-619710.91
5.353499	896.99	4180.74	3317.13	33.37	-14133.48	14122.32	-619710.91
7.137999	919.20	4158.55	3294.92	55.56	-14133.48	14122.32	-619710.91
8.922499	941.41	4136.36	3272.71	77.75	-14133.48	14122.32	-619710.91
10.706998	963.61	4114.17	3250.50	99.95	-14133.48	14122.32	-619710.91
12.491498	985.82	4091.98	3228.30	122.14	-14133.48	14122.32	-619710.91
14.275998	1008.03	4069.79	3206.09	144.33	-14133.48	14122.32	-619710.91
16.060497	1030.24	4047.60	3183.88	166.52	-14133.48	14122.32	-619710.91
17.844997	1052.45	4025.41	3161.67	188.71	-14133.48	14122.32	-619710.91
19.629496	1074.65	4003.22	3139.46	210.90	-14133.48	14122.32	-619710.91
21.413996	1096.86	3981.03	3117.25	233.09	-14133.48	14122.32	-619710.91
23.198497	1119.07	3958.84	3095.05	255.28	-14133.48	14122.32	-619710.91
24.982996	1141.28	3936.65	3072.84	277.47	-14133.48	14122.32	-619710.91
26.767495	1163.49	3914.46	3050.63	299.66	-14133.48	14122.32	-619710.91
28.551996	1185.70	3892.27	3028.42	321.85	-14133.48	14122.32	-619710.91
30.336495	1207.90	3870.07	3006.21	344.04	-14133.48	14122.32	-619710.91
32.120994	1230.11	3847.88	2984.01	366.23	-14133.48	14122.32	-619710.91
33.905494	1252.32	3825.69	2961.80	388.43	-14133.46	14122.32	-619710.91
35.689994	1274.53	3803.50	2939.59	410.62	-14133.48	14122.32	-619710.91

In the principal coordinate system, the shear force in the jth mode would then be:

$$H^j = -\sum_{k,l} m_{kl}\delta_i^k \ddot{\xi}_l^j$$

But

$$\ddot{\xi}_l^j = \varphi_{lj}\Gamma_j f_j(t) = \varphi_{lj} \sum_{k,l} \varphi_{kj} m_{kl}\delta_i^l f_j(t)$$

so that

$$H^j = -\left(\sum_{k,l} m_{kl}\delta_i^k \varphi_{lj} \sum_{k,l} \varphi_{kj} m_{kl}\delta_i^l\right) f_j(t)$$

$$= -\left(\sum_{k,l} m_{kl}\delta_i^k \varphi_{lj} \sum_{l,k} \varphi_{lj} m_{lk}\delta_i^k\right) f_j(t)$$

$$H^j = -\left(\sum_{k,l} \varphi_{lj} m_{kl}\delta_i^k\right)^2 f_j(t)$$

$$(\text{since } m_{kl} = m_{lk}) \quad (36)$$

By analogy with Eq. 35, the expression

$$M_j = \left(\sum_{k,l=1}^{m} \varphi_{lj} m_{kl}\delta_i^k\right)^2 \quad (37)$$

is called the "modal effective mass" in the jth mode.

EXAMPLE

An example of typical computer results for the seismic response of the primary piping system shown in Fig. 4 resulting from response-spectra inputs is shown in Tables 2 to 7 on pages 311 to 315. (These results were obtained in connection with work sponsored by Gilbert Associates and reported in Ref. 13.) Note that Table 7 gives an example of the computer output for the joint loads and resulting seismic stresses combined in accordance with the requirements of Ref. 14.

It may be of interest to note that the seismic stresses are quite small as compared to the stresses generated by pressure, deadweight, and thermal loads. The seismic stresses are, nevertheless, significant in that the piping system has generally been sized for non-seismic loads, thus leaving only a small margin for seismic loads.

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*These programs are described in the Franklin Institute Research Laboratories brochure *Center for Computer Aided Analysis.*

Insurance Aspects of Irradiated Fuel Shipments

By W. F. Kane*

One of the major areas of concern in the shipment of irradiated fuel is with the nuclear liability and property insurances. At present the domestic shipping of irradiated fuel is relatively free of insurance complications; however, some problems still exist with the eastern railroad carriers.

The insurance aspects involved with the international shipping of irradiated fuel are not as clearly defined. If extension of the Price–Anderson Act occurs, further extension of domestic coverage to international shipments will likely evolve to provide indemnification for international shipments of irradiated fuel.

This article deals with the evolution of the nuclear insurance pools and the extent to which they now provide domestic and international insurance coverage. A discussion of insurance with respect both to some current problem areas and to possible future developments is contained in the latter section of this review.

DEFINITIONS[1-3]

Throughout this article, terms such as *spent fuel* will be used in a manner consistent with their usage by the insurance industry. Included in this section is a listing of these terms and their appropriate definitions as recognized by the insurance industry. The following definitions are applicable:

Bodily injury means bodily injury, sickness, or disease, including death resulting therefrom, sustained by any person.

Property damage means physical injury to, destruction of, or radioactive contamination of, property; loss of use of property so injured, destroyed, or contaminated; and loss of use of property while evacuated or withdrawn from use because of the possibility of such contamination or because of the imminent danger of such contamination.

Source material has the meaning given it in the U. S. Atomic Energy Act of 1954 or in any law amendatory thereof.

Special nuclear material has the meaning given it in the Atomic Energy Act of 1954 or in any law amendatory thereof.

By-product material has the meaning given it in the Atomic Energy Act of 1954 or in any law amendatory thereof.

Spent fuel means any fuel element or fuel component, solid or liquid, which has been used or exposed to irradiation in any nuclear reactor.

Waste means any waste material containing by-product material and resulting from the operation by any person or organization of any nuclear facility included within the definition (below) of *nuclear facility* under paragraph 1 or 2 thereof.

Radioactive isotope means any by-product material except such material contained in spent fuel or waste or discharged or dispensed from any nuclear facility.

Nuclear facility means "the facility" as defined in any Nuclear Energy Liability Policy (Facility Form) or by Mutual Atomic Energy Liability Underwriters. The term also means:

1. Any nuclear reactor.
2. Any equipment or device designed or used for (*a*) separating the isotopes of uranium or plutonium, (*b*) processing or utilizing spent fuel, or (*c*) handling, processing, or packaging waste.

*Nuclear Associates International Corp., Rockville, Md.

3. Any equipment or device used for the processing, fabricating, or alloying of special nuclear material if at any time the total amount of such material in the custody of the insured at the premises where such equipment or device is located consists of or contains more than 25 g of plutonium or ^{233}U, or any combination thereof, or more than 250 g of ^{235}U.

4. Any structure, basin, excavation, premises, or place prepared or used for the storage or disposal of waste.

In addition, *nuclear facility* includes the site on which any of the foregoing is located, all operations conducted on such site, and all premises used for such operations.

Nuclear reactor means any apparatus designed or used to sustain nuclear fission in a self-supporting chain reaction or to contain a critical mass of fissionable material.

Nuclear energy hazard means the radioactive, toxic, explosive, or other hazardous properties of nuclear material.

Nuclear material means source material, special nuclear material, or by-product material.

BACKGROUND

With the passage of the U. S. Atomic Energy Act in 1954, private industry was permitted to be licensed by the U. S. Atomic Energy Commission (AEC) to engage in nuclear energy activities on its own account. One of the major sources of difficulty for private industry was the need for large amounts of liability insurance. Although the requests for nuclear liability insurance were on the order of \$50 million to \$100 million, in that particular time period, a liability-policy limit on the order of \$5 million to \$10 million was considered within the insurance industry to be a rather large amount.

In March 1955, under the auspices of the AEC, an insurance study group, consisting of many of the leading executives from the insurance industry, began to examine these problems and make recommendations. The conclusion reached by this study group was that the most serious insurance problem was in the area of third-party liability insurance. It was further concluded by this group that the only way for private insurance to achieve a capacity sufficient to provide such coverage was through the formation of nuclear property-insurance and liability-insurance pools.

NUCLEAR INSURANCE POOLS

In early 1965, two nuclear liability-insurance pools were formed. The larger of the two pools was composed of capital-stock insurance companies and was designated the Nuclear Energy Liability Insurance Association (NELIA). The smaller mutual-company pool was designated the Mutual Atomic Energy Liability Underwriters (MAELU). In the same year, two nuclear property-insurance pools were formed. The capital-stock insurance companies organized the larger of these pools, and it was designated the Nuclear Property Insurance Association (NEPIA). The smaller mutual-company pool was designated the Mutual Atomic Energy Reinsurance Pool (MAERP).

PRICE–ANDERSON ACT

Although the private insurance companies had been able to offer insurance capacity in unprecedented amounts through the insurance pools, public utilities and private industry engaged in the peaceful uses of atomic energy felt that even greater protection was required. As a result, the Joint Committee on Atomic Energy held hearings in 1956 and 1957 on proposals to provide indemnification bills. Subsequently the Price–Anderson Act was passed in September 1957 as Public Law 256 of the 85th Congress, thus providing greater protection for the public and for private industry.

Indemnification is provided under the Price–Anderson Act for any nuclear energy hazard that arises from designated licensed or contractual activities. Such designated activities would include nuclear energy hazards occurring at the site of such activities and those nuclear energy hazards which may occur in the course of transportation to or from the aforementioned activities. Coverage for the transporter under the Price–Anderson Act applies only to those nuclear energy hazards which may occur within the territorial limits of the United States. However, in the event of an indemnified nuclear energy hazard occurring within the territorial limits of the United States, with resultant damages occurring outside the United States, indemnification is also provided by the Price–Anderson Act. In the event that a transporter is provided with an AEC indemnification agreement, his coverage beyond the territorial limits of the United States extends up to \$100 million under the Price–Anderson Act.

The Price–Anderson Act provides indemnification coverage for each nuclear energy hazard over and above the financial protection available from the nuclear insurance pools. This indemnity extends to the licensee

and to anyone else who may be liable for a nuclear energy hazard.

DOMESTIC LIABILITY COVERAGE AVAILABLE FROM INSURANCE POOLS

In their domestic nuclear energy liability-insurance programs, NELIA and MAELU have two types of available policies directly related to the transportation of spent fuel. NELIA and MAELU use the same nuclear energy liability-policy form and endorsements, which have been adopted as standard provisions forms by the National Bureau of Casualty Underwriters and the Mutual Insurance Rating Bureau. The Facility Form[1] is issued to the operators of nuclear facilities and will include, in addition to the operator, coverage for truckers, railroads, airlines, steamship companies, warehousemen, stevedores, and anyone else other than the U. S. Government or any agency of the government who may be liable for bodily injury or property damage resulting from the nuclear energy hazard. The Facility Form policy is continuous and provides a single aggregate limit of liability. It covers the nuclear energy hazard, includes protection at the facility site, and, in many cases, includes handling and temporary storage during the course of transportation. The Facility Form is applicable only within the territorial limits of the United States, its territories or possessions, Puerto Rico, and the Canal Zone.

The Facility Form is tailored to provide broad transportation coverage to those risks which require financial protection under the Price–Anderson Act. The policy has been written in order that only one facility policy will apply to the hazards resulting from a single shipment of irradiated fuel. The Facility Form policy issued to the operator of an indemnified nuclear facility is written to provide coverage for shipments to any other location including indemnified nuclear facilities, and coverage from any other location other than indemnified nuclear facilities.

The Facility Form policy can also be issued to a nuclear facility that is not required to furnish financial protection under the Price–Anderson Act. Then coverage by the Facility Form is provided for shipments to that facility from one owned by the United States and from that facility to any other location except to an indemnified nuclear facility. The Facility Form policy defines an indemnified nuclear facility as a facility that is required to furnish financial protection under the Price–Anderson Act.

Thus a transporter is covered by the Facility Form policy of an indemnified nuclear facility for shipments to any location except to another pool-insured indemnified nuclear facility. In this case the transporter is afforded coverage under the latter's Facility Form policy. For shipments from a pool-insured *nonindemnified* nuclear facility, the transporter is covered by the Facility Form policy of that facility except in instances where the shipment is made to a pool-insured *indemnified* nuclear facility, in which case he is covered by the latter's policy. The transporter is also covered for those shipments to any pool-insured facility from a government-owned facility.

Two additional features of the Facility Form are of particular interest to the transporter of irradiated fuel. Third-party liability-insurance coverage for damage to companion cargo and to the transporting conveyance is covered if the nuclear energy hazard occurs in the course of transportation covered by the Facility Form. Liability for damage to the irradiated fuel would, however, be excluded from coverage by the Facility Form. Coverage for loss or damage to the irradiated fuel will be discussed later.

In addition to the coverage provided under the Facility Form, the transporter may also purchase a Nuclear Energy Liability Policy (Supplier's and Transporter's Form). This form[2] makes available excess liability coverage over and above that which is provided by the Facility Form. Unlike the Facility Form, the Supplier's and Transporter's Form does not provide derivative coverage. The liability coverage provided by the Supplier's and Transporter's Form extends to the same class of insured as the general liability policies. This would include officers, directors, stockholders, and partners and, in addition, would cover employees acting within the scope of their duties. It is also important to note that the Supplier's and Transporter's Form is strictly a third-party liability contract. It does not provide coverage of off-site property damage (to the transporter's property), as in the case of the Facility Form. It is impossible, therefore, for a transporter of spent fuel to obtain specific insurance in his own name from the liability-insurance pools to protect him from damage to his own property.

DOMESTIC PROPERTY COVERAGE AVAILABLE FROM INSURANCE POOLS

There are two types of domestic nuclear property-insurance policies which are available from NEPIA and MAERP and which are directly related to the transportation of spent fuel. Identical policy forms are issued by each of these pools.

The Shippers Transportation Policy[3] indemnifies the shipper and legal representatives to the extent of the cash value of the property at the time of loss, but not exceeding the amount it would cost to repair or replace the property. The policy covers the spent fuel, as well as the containers, materials, and supplies shipped with the fuel. The policy applies only to shipments between points within the 48 contiguous states of the United States, Alaska, and the District of Columbia or shipments to Canada which originate within the continental United States. Shipments via the Panama Canal and waterborne shipments to and from Alaska or Canada are excluded. The policy covers the spent fuel from the time it leaves the premises until it is delivered to the destination.

The Carriers Transportation Policy[4] indemnifies the carrier and legal representatives to the extent of the carrier's liability imposed by law upon the carrier under bill of lading, shipping receipt, or contract. All risks of direct physical loss of the spent fuel and containers, including general average and salvage charges, are indemnified. The restriction regarding geographical limitations and the duration of coverage are the same as those for the Shippers Transportation Policy.

GOVERNMENT INDEMNITY

Transporters may also be protected under an agreement of indemnification issued by the AEC pursuant to the Price–Anderson Act. The Act permits the AEC to issue contracts of indemnification to its contractors and provides that such contracts shall extend to anyone else who may be liable with respect to nuclear energy hazards occurring from related contractual activities. Where the AEC furnishes such protection, payment is made from the first dollar of loss.

Under the private-licensee program, the AEC requires licensees of operating reactors, critical facilities, and other production or utilization facilities to furnish financial protection in the form of private insurance or self-insurance, or a combination of such measures, and will provide indemnity up to $500 million in excess of the required financial protection. This indemnification extends to anyone who may be liable, including the transporter.

REGULAR LIABILITY POLICIES

In addition to those liability-insurance policies which are available from NELIA and MAELU, certain protection is also available to the transporter of spent fuel against the nuclear energy hazard under regular automobile and general liability policies issued by U. S. casualty carriers. Certain liability policies, as will be discussed later, do not carry a nuclear liability exclusion clause; therefore, coverage is provided for liability caused by the nuclear energy hazard in the same manner and to the same extent as any other hazard is insured by that policy.

LIABILITY POLICIES NOT SUBJECT TO NUCLEAR EXCLUSION

There are certain policies which, although purchased by interests that are directly or indirectly engaged in the transportation of spent fuel, may not carry a nuclear exclusion clause. Examples of some of these policies include the following:

1. Protection and indemnity policies, when issued to shipowners by the American ocean-marine market.
2. Hull policies, which provide third-party liability-insurance protection to shipowners under the collision clause for damage to another ship or vessel and associated cargo.
3. Stevedores' legal-liability policies, which afford property-damage liability insurance.

In general, cargo policies do not carry third-party liability-insurance protection. Shippers may secure product liability insurance for their third-party liability protection from incidents arising out of the transportation or use of their products by others away from their premises.

INSURANCE-RELATED DOMESTIC SHIPPING PROBLEMS

Although Price–Anderson coverage makes most shipments of spent fuel within the United States a great deal easier, there are still difficulties involving the eastern railroads.[5] This system extends generally east of the Illinois–Indiana state line and north of the Ohio River. The eastern railroads do not represent themselves as transporters of spent nuclear fuel in common carriage. They have transported spent fuel, however, for the private sector of the nuclear industry under specially negotiated contracts. Shipments for the U. S. Government and its contractors are generally made under a special provision of the Interstate Commerce Act. The level of freight rates assessed in both instances generally has been the same.

The major insurance protection now available to the rail carriers (as well as motor and air carriers)

results from the Facility Form policies of NELIA and MAELU. Since, as mentioned previously, this policy is derivative, it extends to the railroad carriers. Additional direct insurance is also available through the Supplier's and Transporter's Policy, which is also issued by NELIA and MAELU.

One of the primary aspects concerning the railroad carriers is the possible personal-injury claims made against them based on alleged continued exposure to radiation rather than as the direct result of some specific incident. The Facility Form of NELIA and MAELU excludes coverage for the continued-exposure condition. The railroad carriers feel that, if several shipments were relied upon as the bases for the cause of a claimant's injury, it would not be possible to relate such an injury to a specific shipment. If this could not be done, then it would not be possible to relate the injury to a specific insurance policy. In the event that a specific policy could be identified, they feel that it might not cover the claim in full, and possibly not at all. The latter supposition is based on a clause contained in the Facility Form which requires claims to be made prior to 2 years after cancellation of the insurance policy. The policy may also be canceled by the pool by mailing to the insured and to the AEC written notice stating when, not less than 90 days thereafter, such cancellation shall be effective. Claims of the continued-exposure type, it is reasoned by the railroad carriers, may not even be known to exist until after a policy of this type could be terminated.

Additional reservations have been expressed by the railroad carriers and deal primarily with interpretations of the pool policies and the Price–Anderson Act indemnification.[6]

INTERNATIONAL SHIPMENTS

As noted before, liability-insurance coverage is not available to cover the transportation of spent nuclear fuel beyond the territorial limits of the United States, under the NELIA and MAELU domestic insurance programs. To provide liability coverage in this area, NELIA and MAELU have each developed independent foreign programs.

The Nuclear Energy Liability Foreign Coverage Policy (Supplier's and Transporter's Form)[7] offered by NELIA is very similar to their domestic Supplier's and Transporter's Form. The limit-of-liability coverage provided by the policy covering foreign shipments is $10 million. A somewhat similar policy is available from MAELU and is limited to $5 million. Additional coverage from overseas insurance companies in the amount of $5 million can be obtained.

If, however, a shipment of spent fuel is made to a European facility indemnified under the "Paris Convention," $100 million indemnification is provided. The Paris Convention is an indemnification arrangement similar to that provided under the Price–Anderson Act for domestic shipments. A proposed expansion of the Price–Anderson Act to cover foreign shipments of spent nuclear fuel is discussed in the next section.

Regarding nuclear property insurance, there are no special coverages similar to NEPIA or MAERP which provide coverage outside the United States. In ocean transport, the carrier normally limits his cargo insurance to $500 per package; therefore additional cargo insurance is required. Typical cargo insurance would cost on the order of 15¢ to 20¢ per $100 of evaluation, depending upon the age and condition of the ship. The amount of cargo insurance obtained should be sufficient to cover the spent nuclear fuel, shipping container, and freight cost.

PROPOSED REVISION TO PRICE–ANDERSON ACT

Legislation has been proposed to amend the Atomic Energy Act of 1954 to extend indemnification to international shipments of nuclear materials. The proposed new language would broaden the definition of nuclear incidents to mean those occurrences which arise out of the transportation of source, special nuclear, or by-product material on vessels of U. S. registry and which take place in the course of such transportation and outside the territorial limits of the United States or any other nation. This legislation would thus provide coverage on the high seas or within a zone, such as Antarctica, which does not belong to a particular government. As before, if an occurrence is to qualify as a nuclear incident, it must cause, within or outside the United States, injury to the body, sickness, disease, death, loss of property, or damage to property arising out of or resulting from the radioactive, toxic, explosive, or other hazardous properties of the source, special nuclear, or by-product material.

Those persons who would be indemnified under the proposed legislation are defined. With respect to international shipments, the term "person indemnified" is to be limited to the person with whom an indemnity agreement is executed (licensee or contractor) and to any other person who may be liable for public liability by reason of his activities within the

United States or on board the vessel transporting the nuclear material while the vessel is on the high seas or who may be liable in connection with his ownership or operation of the vessel. Also indemnified would be any other person who has his principal place of business in the United States or who is the agent or employee of such other person and who may be liable for public liability. Specific types of individuals or agencies that would be covered by this definition are the licensee, the contractor, the U. S. cask designer and/or manufacturer, the stevedores in the American port, the seamen on board the ship, the owner or master of the vessel, and any employee of the American cask manufacturer responsible for supervising unloading of the cask at a foreign port.

Coverage for international shipments would not include the vessel transporting the nuclear material and its equipment, fuel, and stores, which would be covered by hull insurance. In addition, no coverage of the nuclear material or the shipping container would be available via this legislation.

The maximum liability coverage available for international shipments would be $100 million, together with the amount of financial protection through commercial insurance for nuclear incidents. The aggregate liability would be limited to $115 million.

A new subsection would authorize the AEC, until Aug. 1, 1977, to enter into additional agreements of indemnification with its licensees and contractors who are engaged in activities in the United States and who are already covered by an indemnity agreement. Such additional agreements would indemnify the licensee, contractor, and other indemnified persons for public liability, in excess of the level of financial protection required, resulting from nuclear incidents which arise from transporting nuclear material to or from a U. S. plant and which take place during the transport of the nuclear material outside the territorial limits of the United States or any other nation.

In the AEC's analysis of the proposed legislation to amend Section 170 of the Atomic Energy Act, the following passage is excerpted:[8]

> Following the pattern of domestic arrangements, with respect to licensees the Commission must, and with respect to contractors may, require that financial protection be provided. The amount of financial protection to be required would be $15 million (the amount which appears to be presently available). The Commission may establish a greater or lesser amount of financial protection taking into consideration the amount of liability insurance available from private sources (domestic and foreign), and the cost and terms of such insurance.
>
> The amount of indemnity for persons indemnified in connection with each nuclear incident would be $100 million including the reasonable costs of investigating and settling claims and defending suits for damage.
>
> The amount of indemnity would be reduced by the amount that the financial protection required exceeds $15 million.

CONCLUSIONS

In the domestic shipment of spent nuclear fuel, shippers and transporters have considerable protection against the nuclear energy hazard under the NELIA–MAELU policies and government indemnity provided by the Price–Anderson Act. Should the shipper or transporter require additional third-party liability protection, supplier's and transporter's policies may be purchased from the NELIA and MAELU pools. Additional coverage is available to the shippers and transporters through their regular liability policies.

Where third-party liability insurance is required for international shipments, the NELIA and MAELU pools offer independent policies. Additional protection may also be obtained from overseas insurance pools. Depending on the shipping destination, indemnification under the Paris Convention may also be available. U. S. legislation to expand the Price–Anderson Act to cover international shipments will provide additional protection in this area.

In the area of property insurance, adequate domestic coverage is available under the NEPIA and MAERP policies. For international shipments, the NEPIA and MAERP policies do not provide coverage; however, the required coverage can normally be obtained through marine cargo insurance.

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The BN-350 Reactor

By Walter Mitchell III*

The efficient utilization of nuclear fuel is vital if power reactors are to fulfill the role assigned them in meeting future requirements for abundant electric power. The consensus is that the development and commercial use of fast breeder reactors is necessary if we are to manage our nuclear fuel resources wisely. However, there are divergent opinions on the course that fast breeder development should follow, and a comparison of two national programs can be useful in noting some of the differences in approach.

The liquid-metal-cooled fast breeder reactor (LMFBR) program in the United States is following a detailed, carefully conceived plan to develop or select the materials, components, systems, and techniques that are needed for a large, sophisticated, efficient, sodium-cooled breeder. Construction of such a reactor is deferred in favor of component demonstration. In contrast, the breeder program in the Soviet Union is characterized by the construction of larger reactors—most notably the BN-350, a 1000-Mw(t) sodium-cooled breeder nearing completion at Shevchenko on the Caspian Sea.

The reader will have no difficulty keeping abreast of progress in the LMFBR program of the United States and assessing the effectiveness of that approach to breeder development. For a basis of comparison, it seems worthwhile to present here some of the design details of the BN-350, as can best be determined by a review of the available information on that plant. When additional information becomes available on plant design and, ultimately, operation, the degree of success of the Russian approach to development should be reasonably clear.

*Southern Nuclear Engineering, Inc., Dunedin, Fla.

Two notes are in order before beginning the description of the BN-350. First, certain design features of the reactor have changed during the 5-year period since its announcement. The parameters and features given here are thought to reflect the most current design for which data are available. Second, metric units are used throughout this article. It is deemed undesirable to effect the rounding-off that would be necessary to present dimensions, performance figures, etc., in units which may be more comfortable for the reader.

GENERAL PLANT FEATURES

The BN-350 gets its name from the facts that it is a fast power reactor—that accounts for the BN portion of its name[1]—and that, with the use of a standard condensing turbine, the electric-power output of the station[2] would be 350 Mw. The BN-350 is not a strict power-production reactor, however. It is a dual-purpose, electric generation-water desalination plant which will use a backpressure turbine to produce 150 Mw of electric power. The system also furnishes process steam for a desalination plant which produces 120,000 m^3 of fresh water per day.

Within the framework of a discussion of fast breeder reactors, the BN-350 cannot be said to be a very sophisticated system. It is based in large measure on the experience obtained with the 5-Mw(t) BR-5 sodium-cooled test reactor which began operation in the USSR in 1959. Some of the basic parameters (sodium outlet temperature from the reactor, core power density, burnup, etc.) are based on BR-5 values. It appears that the basic design of the BN-350 was developed during the period 1961-1963, and the initial

goals set for the reactor are reasonably modest when viewed from the U. S. position some years later. The Russians will build other, larger breeders (the BN–600 is said to be under construction at the present time[3]), and it is said[4] of the BN–350 plant: "The atomic electric power station is both productive and experimental in nature. It is designed for the development and mastery of the reactor type used. . . ."

The plant site is about 3.5 km from the seashore and has as its principal structures the reactor building, a turbine building, a distillation complex, and a building that houses a conventional thermal power plant.[5] The conventional thermal plant, with a capacity of around 60 Mw, is being built because the electric–power output of the BN–350 is insufficient to support the new industrial complex being developed in the area. The reactor building is just that; a conventional industrial building similar to those which house other Russian reactors and not a containment structure. The principal section of the building is 110 m long, 45 m wide, and about 60 m high. Annexes to the main section house electrical controls, personnel facilities, fuel–storage ponds, a ventilation center, etc. A tall stack is used in conjunction with the ventilation center for the discharge of low–activity air to the atmosphere.[6] It is interesting to note that a 150 to 200 ton/hr "experimental–industrial evaporation unit" was built to supply fresh water to the city of Shevchenko prior to the start of construction of the BN–350. The distillation unit was designed and built in 1962 and 1963, with operation commencing[7] in October 1963.

Figure 1 is a simplified diagram showing flows, temperatures, and pressures in the BN–350. A three–circuit scheme is used for transferring heat from the reactor to the steam–water system. The Russians say[8] that the pot–type arrangement, in which all primary–loop equipment is placed in a common tank filled with sodium (as in EBR–II, Britain's PFR, and the French Phenix reactor), seems to be more promising than the loop type, in which the reactor, pumps, and heat exchangers are placed in separate compartments and connected by pipes (as in Fermi and the small Dounreay reactor). Nevertheless, the BN–350 is a loop–type plant. The primary circuit is provided with six parallel loops, as is the intermediate circuit. During normal operation, five loops are used, with the sixth loop in each circuit acting as a standby.[9] The sodium of the loops of the first circuit transfers heat from the reactor to the coolant (also sodium) of the intermediate circuit. After leaving the heat exchangers, the sodium in the intermediate circuit passes to the steam generators, where it gives up its heat to the steam–water circuit. Three turbine generators are used, each having a rated power of 50 Mw. The exhaust steam from the turbines is used in the desalination complex; it is returned as water to the steam–water system shown in Fig. 1.

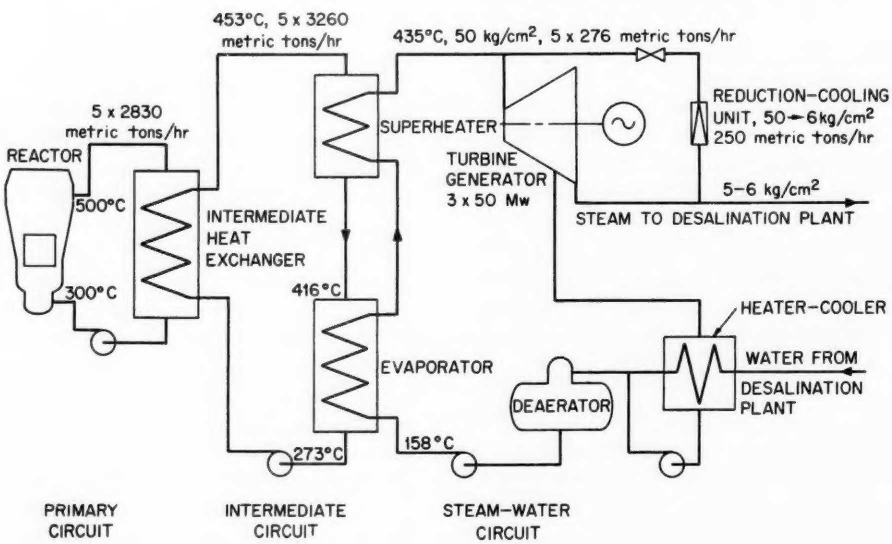

Fig. 1 Simplified flow diagram of the BN-350 plant.

Some basic parameters of the installation are tabulated below.

Reactor thermal power, Mw	1000
Electrical output, Mw	150
Freshwater output, metric tons/day	120,000
Start of construction	October 1964
Date critical (anticipated)	1971–1972
Primary coolant	Sodium
Intermediate coolant	Sodium
Reactor cooling-circuit arrangement	Loop type
Number of primary and intermediate loops	
Required (full power)	5
Standby	1
Installed	6
Containment building	None

REACTOR VESSEL

The BN-350 core, blanket, refueling mechanism, and other reactor components are enclosed in a sodium-filled vessel, as shown in Fig. 2. The vessel is made of 18 Cr–9 Ni stainless steel (a type 304 stainless) and varies in diameter from a maximum of 6 m to a minimum of 2.2 m. Wall thickness is 30 mm, and overall height of the vessel is 13 m. The vessel is field assembled at the reactor site from nine large subassemblies that are "factory-made parts, completely finished, adjusted to fit each other, and prepared for assembly and welding. . . ."[5] Each part is rail transportable.

Sodium of the primary circuit enters the reactor vessel near its bottom through the six pipes of the six primary loops (five in use during operation). The

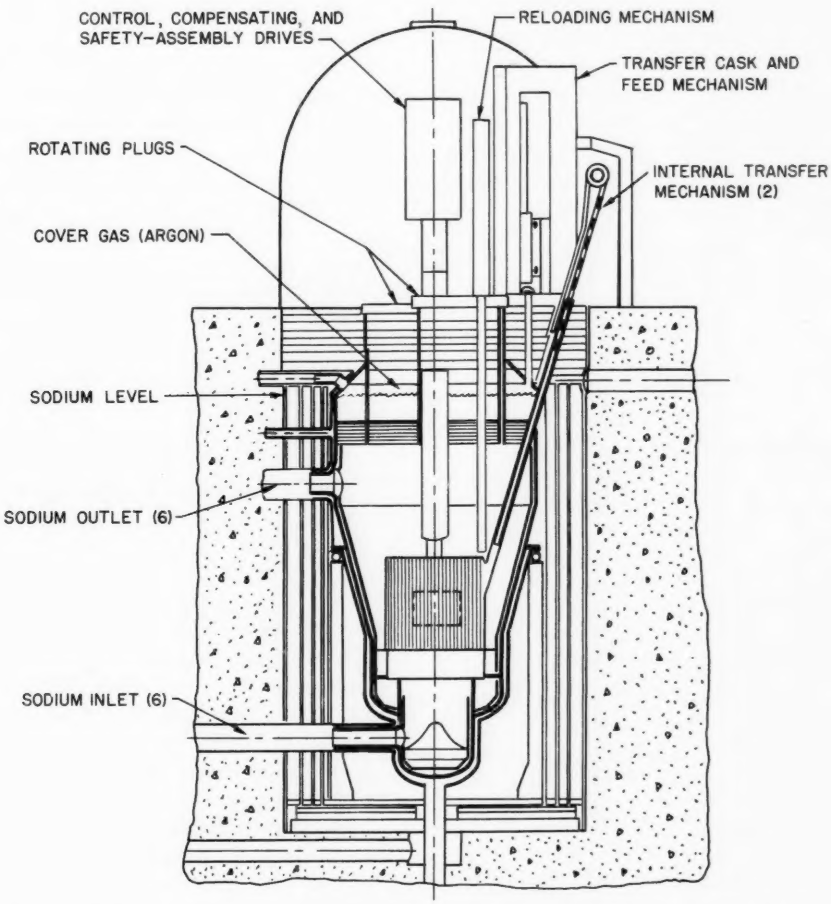

Fig. 2 The BN-350 reactor.

sodium leaves the 500-mm-diameter inlet pipes and flows upward through the core and blanket. After mixing in the region of the vessel above the core and blanket, the sodium leaves the vessel through five of the six 600-mm-diameter outlet pipes of the primary circuit.

Inner flanges in the reactor vessel serve as the support for a header upon which the core and blanket are mounted. The reactor vessel itself is supported from a flange at approximately its mid-height. The vessel is protected with a thermal shield consisting of stainless steel 60 mm thick. Two percent of the total sodium flow through the reactor vessel is diverted to the annular gap formed by the inner wall of the vessel and the outer surface of the thermal shield. In this manner the operating temperature of the vessel is reduced from 500°C (sodium outlet temperature) to 420°C.

Internal radiation shielding is provided around the blanket and in the upper region of the vessel. The upper portion of the reactor vessel serves as a support for two rotating shield plugs—a large one and the usual smaller, offset plug. A secondary sodium containment, consisting of a 10-mm-thick envelope around the reactor vessel, is provided to prevent a major loss of sodium in the event of a leak in the reactor vessel proper.

Vessel diameter, m	
Maximum	6.0
Minimum	2.2
Overall height, m	13.0
Wall thickness, mm	30
Material	18 Cr–9 Ni stainless steel
Loop pipes, number and diameter, mm	
Inlet	6 X 500
Outlet	6 X 600
Thermal shield	
Material	Stainless steel
Thickness, mm	60
Vessel operating temperature, °C	420
Maximum pressure in primary circuit, kg/cm^2	10
Leak protection	Secondary containment, 10 mm thick
Support	Mid-height flange and rollers

CORE AND BLANKET

The core and blanket are formed by vertical assemblies with a hexagonal cross section. The elements measure 96 mm across the flats of the hexagon and are spaced 98 mm on centers.[9] The core contains 211 positions, of which 199 are occupied by fuel assemblies. The remaining 12 positions are for control, compensating, and safety elements.[10] The core is 1.06 m in height and has an effective diameter of 1.495 m. The power released in the core is 890 Mw, giving an average power density of around 480 kw/liter. A report on heat-transfer experiments relative to the core and blanket elements of the BN-350 was presented at the 1965 Detroit meeting on fast reactor technology.[11] The fuel has been reported to be mixed Pu-U oxides or enriched uranium oxide,[12] although the possibility of using the enriched UO_2 as the first charge has been mentioned since the earliest announcements of the reactor.[13-15]

The core consists of two regions: (1) a central region containing 109 assemblies and (2) a peripheral region with 90 assemblies. The elements for the two regions differ with respect to fuel enrichment in the core and also with respect to end-fitting design.

The core is surrounded by 600-mm-thick radial and axial blankets. The core assemblies, which are 3.5 m in overall length, contain blanket sections at each end of the core region. The radial blanket, which surrounds the core, is made up of 440 assemblies of the same size as those which form the core and axial blankets. The radial blanket is divided into two separate regions: an inner zone with 120 assemblies, and an outer zone with 320 assemblies. Elements of the inner zone of the radial blanket are provided with end fittings of the same type as those of the peripheral core elements. These outer core and inner blanket elements are said[10] to be interchangeable to provide a means by which the critical size of the core can be adjusted. Orifices in the bottom fittings of the assemblies, as well as in the support grid in which they fit, are used to proportion the sodium flow.

The possible PuO^2–UO^2 core has been reported[12,15] to be 19% PuO_2, or 15% PuO_2 in the first zone and 20% PuO_2 in the second zone.[16] There is also some slight discrepancy in the reporting of the enrichment of the most likely first core—the enriched UO_2 one. References 12, 15, and others list the enrichment as 23%, whereas it is given as 17% in the first zone and 26% in the second zone in Ref. 16. Apparently, the reported single value is merely a simple average of the enrichments in the two core zones.

The physical arrangement of the core assemblies is one in which small-diameter, nonvented fuel rods are enclosed within the hexagonal outer sheath of the assembly, with larger-diameter axial blanket rods above and below. In the radial blanket assemblies,

full-length, large-diameter rods are spaced within the hexagonal sheath. Details are given in the tabulated data below.

The choice of cladding material is not clear, other than that it is said to be stainless steel. Some special alloys are undergoing development specifically for use as cladding in the BN-350; one which has been discussed[17] and which is particularly interesting is produced in the basic composition 16 Cr-15 Ni-3 Mo-Nb as well as in a variation with 0.005% boron added. Russian comments on the effects of the boron addition to this material, as well as others, are surprising:

> The irradiation (to about 10^{21} nvt) of steels of all classes containing boron in the amount 0.005% showed that this amount of boron not only does not lead to a still greater reduction in the high-temperature plasticity of steels than in steels without boron, but, as a rule, improves it in comparison with steels in which boron is specially not introduced.[18]

The breeding ratio for the reactor using enriched-uranium fuel is reported[19] to be 1.16, whereas that for the plutonium-fueled core is given as 1.5.

Reactor thermal power, Mw	1000
Power released in core, Mw	890
Core height, m	1.06
Effective diameter, m	1.495
Core volume, liters	1870
Power density, kw/liter	
Average	480
Maximum	840
Maximum cladding-surface temperature, including all hot-spot factors, °C	695
Fuel material	UO_2 or $PuO_2 + UO_2$
Critical mass	
^{235}U, kg	1040
^{239}Pu, kg	850
Core loading at rated power	
^{235}U, kg	1170
^{239}Pu, kg	940
Maximum burnup, %	5
Breeding ratio	
^{235}U fuel	1.16
^{239}Pu fuel	1.50
Neutron flux, maximum, neutrons/(cm^2)(sec)	1×10^{16}
Neutron energy, kev	150 to 200
Neutron lifetime (^{235}U fuel), sec	4.5×10^{-5}
Blanket thickness, mm	
Axial, top and bottom	600
Radial	600
Fertile material	Depleted UO_2
Core composition, %	
Fuel	44.7
Sodium	31.1
Steel	24.2
Axial blanket composition, %	
Fertile material	43.8
Sodium	41.9
Steel	14.3
Radial blanket composition, %	
Fertile material	60.8
Sodium	21.6
Steel	17.6
Core and blanket assemblies	
Cross-section shape	Hexagonal
Distance across hex flats, mm	96
Hex wall thickness, mm	2
Overall height, m	3.5
Center-to-center spacing, mm	98
Structural material	Stainless steel
Number in core	
Inner zone	109
Outer zone	90
Number in radial blanket	
Inner zone	120
Outer zone	320
Fuel elements	
Type	Sealed tubes containing fuel pellets
Number per assembly	169
Cladding material	Stainless steel
Length, m	1.06
Outside diameter, mm	6.1
Cladding thickness, mm	0.35
Pitch (triangular), mm	7.0
Fuel form	Oxide pellets
^{235}U enrichment for UO_2 core, %	
Inner zone	17
Outer zone	26
^{239}Pu content for PuO_2-UO_2 core, %	
Inner zone	15
Outer zone	20
Effective oxide density in UO_2 core with respect to available volume for fuel, g/cm^3	8.0
Axial blanket elements	
Type	Sealed tubes containing depleted UO_2 pellets
Number per assembly	37 top, 37 bottom
Cladding material	Stainless steel
Outside diameter, mm	12.0
Cladding thickness, mm	0.4
Spacing, mm	14.5
Smeared density of UO_2, g/cm^3	9.5
Radial blanket elements	
Type	Sealed tubes containing depleted UO_2 pellets
Number per assembly	37
Cladding material	Stainless steel
Outside diameter, mm	14.2
Cladding thickness, mm	0.5
Spacing, mm	14.8

CONTROL AND SAFETY

The BN-350 plant has an overall centralized control system.[20] Safety, regulation, and control of the reactor are tasks of the centralized system, which consists of the following subsystems:

1. A system for measuring and controlling the reactor power and period (including subcritical range).
2. An automatic control system.
3. A reactivity-change compensating system.
4. A reactor safety system.

There are 12 compensating, control, and safety elements; these elements move in the 12 lattice positions within the core not occupied by fuel assemblies.

The automatic control system is said[10] to maintain a selected power level in the range from 0.1 to 100% power. The system is also capable of performing automatic startup of the reactor when used in conjunction with a reactor power monitor and the reactivity-change compensating system. Power-level changes are made at preset rates in the range between 1.0 and 100% power.

The automatic control system contains two assemblies that can move vertically in 2 of the 12 core lattice positions provided for the control, compensating, and safety systems. The automatic control assemblies are independent and redundant, with one used for operation and one in standby. The reactivity-change compensating system contains seven assemblies, and the remaining three lattice positions are allocated to the three assemblies of the reactor safety (scram) system.[21]

The drive systems for the control assemblies, the compensating assemblies, and the scram assemblies are basically the same. The assemblies are moved by motors with reduction gears and rack-and-pinion drives. The scram-assembly drive mechanisms have, in addition, one accelerating and one shock-absorbing spring. The accelerating spring has a force of 100 kg.

The automatic control assemblies consist of multiple absorber rods containing boron carbide. Each rod contains a gas cavity in its upper part to collect the helium formed as a result of burnup of the boron.

Number of assemblies in automatic control system	2
Reactivity worth of each assembly, %	0.20
Assembly travel, mm	750
Speed of movement, mm/sec	
Automatic mode	150
Manual mode	10
Absorber	
Material	B_4C enriched to 80% in ^{10}B
Form	Rods
Rod diameter, mm	9.6
Number of rods per assembly	7

The compensating system accommodates temperature effects and burnup. During the approach to a selected power level under automatic control, the compensating system also adjusts the position of the automatic control rods. Burnup compensation is accomplished by six of the seven assemblies in the compensating system. These are fueled assemblies that contain an active section consisting of fuel elements similar in design to the core elements. The seventh assembly in the system is a temperature compensator containing boron carbide absorber rods.

The period of reactor operation between refuelings depends upon the effectiveness of the compensating system and the rate of reactivity change caused by burnup. One month of operation results in the following reactivity changes:

	UO_2 fuel	PuO_2-UO_2 fuel
^{235}U depletion	−0.0135	
^{239}Pu depletion and buildup	+0.0087	−0.0056
^{238}U depletion	+0.00057	+0.00089
Fission-product buildup	−0.0012	−0.0014
Overall change	−0.0054	−0.0061

The most recent available information indicates that the value of the burnup compensators is sufficient to provide an operating period of 50 days.

Calculated values of temperature and power coefficients of reactivity—not in very good agreement—are given in Refs. 10, 12, and 16. Details of the reactivity-change compensating system are listed below.

Number of assemblies in compensating system	7
Fueled assemblies (burnup compensators)	
Number	6
Weight of uranium per assembly, kg	18
Reactivity worth of 6 assemblies, %	1.1
Form	Similar to core fuel assemblies
Vertical movement, m	1.06

Absorber assembly (temperature compensator)	
Number	1
Reactivity worth, %	1.0
Poison	
Material	B_4C
Form	Rods
Number of rods	31
Speed of movement, mm/sec	10

The three movable assemblies of the safety (scram) system are absorbers. Boron carbide is the poison material in the safety assemblies, as indicated in the following tabulation.

Number of assemblies in safety system	3
Total reactivity worth, %	3.5
Assembly travel, m	1.26
Insertion time, sec	0.7
Speed of movement, raising, mm/sec	5
Absorber	
Material	B_4C
Form	Rods
Number of rods per assembly	7

The control-assembly mechanisms have been subjected to extensive tests in hot sodium loops.[20,21] The duration of the hot tests was greater than 8000 hr during which time the automatic control mechanism underwent 20,000 insertion-withdrawal cycles, the compensating mechanism 2500 insertion-withdrawal operations, and the safety mechanism 1600 scrams and 400 slow insertion-withdrawal cycles. The mechanisms functioned properly, and there were no breakdowns.

Monitoring and measuring instruments for fast reactors are described in Ref. 22. The parameters to be monitored are conditionally divided into three groups, defined as follows:

> First group—parameters for the reactor and for technological circuits and equipment directly related to the reactor and affecting its operation (first, second and third circuits, pumps, steam generators, cooling systems for the shielding, pumps and flow control elements and so on);
>
> Second group—parameters for the circuits and equipment related to operation of the basic circuits but having no direct effect on reactor operation (filter traps, oxide indicators, systems for cooling them and so on);
>
> Third group—parameters for independent technological systems not related to the operating reactor and main circuits (systems for treating the coolant, general exchange ventilation, gas equipment, thermal engineering systems for the technological transportation section and so on).

FUEL HANDLING

The reactor refueling system consists of the two previously mentioned rotating plugs (one of which is eccentric), a reloading mechanism, two internal transfer mechanisms, a transfer cask with feed mechanism, two conveyor drums, a cleaning station, and a tank to hold the assemblies that have been removed from the reactor.[9] The design of the refueling-system components permits handling of the safety- and control-system assemblies as well as the core and blanket assemblies.[20,21]

The reloading mechanism is located in the smaller (eccentric) rotating plug and, by proper rotation of the two plugs, can be positioned above any lattice or storage position in the reactor. During reactor operation the rotating plugs are sealed with a Sn-Bi eutectic. The eutectic has a melting point of 138°C and is melted by electric heaters when movement of the rotating plugs is desired. In addition to the freeze seal, a mechanical seal utilizing a rubber ring bolted against the sealing faces is used. During the refueling procedure, the rubber-ring seal is raised slightly so that the plugs can rotate.

A storage pit is located at the periphery of the radial breeder zone in the reactor, and core assemblies that have reached the burnup limit are transferred to this area initially for storage. Fresh core assemblies are installed in their place in the core proper. When the plutonium content of the assemblies in the radial breeder zone reaches a predetermined value, the assemblies are removed directly from the reactor vessel, with no internal storage. The assembly to be removed from the reactor vessel is transferred from the storage pit or from the radial breeder zone to the receptacle of one of the internal transfer mechanisms (see Fig. 2). The reloading mechanism is used for this movement. After receiving the assembly to be discharged, the internal transfer mechanism lifts it to a position where it can be grasped by the feed mechanism of the transfer cask. The feed mechanism moves the assembly into the transfer cask, which then moves to a position above one of the conveyor drums. The assembly is discharged into the conveyor, from which it is moved to a cleaning station and then to a decay tank.

As explained in Ref. 9: "All operations in bringing the reloading mechanism to the given packet, extracting the depleted packet and loading in fresh packets are taken care of automatically by command from a control panel."

All components of the fuel-handling system have been subjected to thorough tests at appropriate temperatures. The best description of the components and tests is given in Ref. 20 (a lesser amount of information can be found in Ref. 21).

PRIMARY CIRCUIT

As mentioned above, six parallel primary loops are provided to remove the heat from the reactor. Five of the loops are used during full-power operation, and the sixth is held in reserve. Each primary-circuit loop contains an intermediate heat exchanger, a pump, piping, a check valve, and two gate valves with freeze seals to isolate the circuit from the reactor. The primary circuit is also supplied with a number of auxiliary components; the most important of these are drain tanks and cold traps.

At full power, sodium flows through the reactor at the rate of 14,150 metric tons/hr; the sodium temperature is raised from 300 to 500°C. The five operating loops of the primary circuit, each carrying 2830 tons/hr, then transport the hot sodium to the intermediate heat exchangers, where it gives up its heat and raises the temperature of the sodium of the intermediate circuit from 273 to 453°C. Pumps (one per loop) return the primary-circuit sodium to the reactor for another cycle.

The intermediate heat exchanger for each of the six loops of the primary circuit consists of two sections connected in parallel, as shown in Fig. 3. A section consists of a horizontal tank in which three heat-transfer bundles are immersed; each bundle contains 343 U-tubes.[23-25] The bundles in each section are connected in series with respect to the sodium in both the primary and intermediate circuits. As shown in Fig. 3, a smoothing vane assembly is installed in the shell (primary) side before the first tube bundle to provide uniform flow of the primary sodium. The basic characteristics of the heat exchanger are as follows:

Thermal-power capacity, Mw	200
Primary-circuit sodium temperature, °C	
At heat-exchanger inlet	500
At heat-exchanger outlet	300
Primary-circuit sodium flow per loop, metric tons/hr	2830
Intermediate-circuit sodium temperature, °C	
At heat-exchanger inlet	273
At heat-exchanger outlet	453
Intermediate-circuit sodium flow per loop, metric tons/hr	3260
Heat-exchange surface, m^2	1120
Pressure drop, kg/cm^2	
Primary circuit	0.146
Intermediate circuit	2.24
Number of U-tubes per bundle	343
U-tube dimensions, mm	
Diameter	28
Wall thickness	2
Spacing (square lattice)	35
Material	18 Cr–9 Ni stainless steel (304)

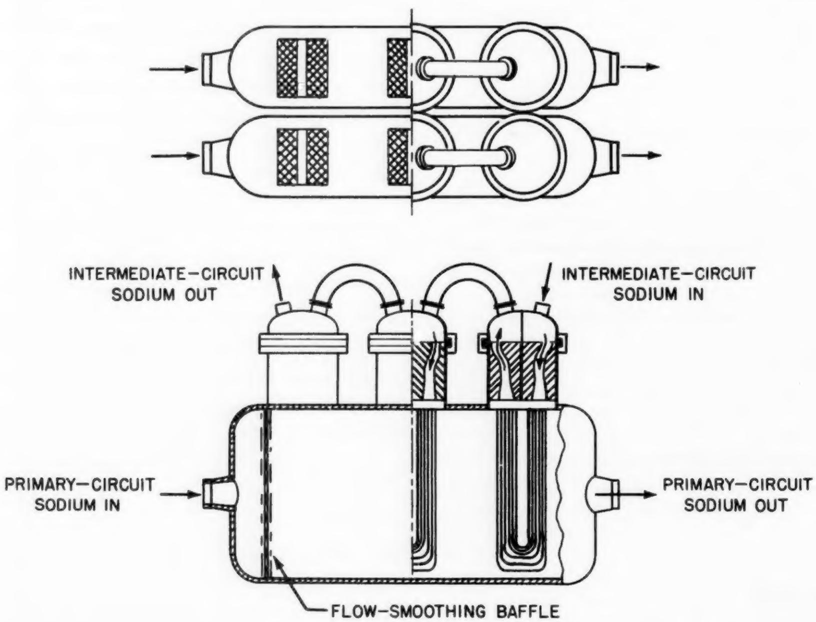

Fig. 3 Sodium–sodium heat exchanger for the BN-350.

The pumps of the primary circuit are sealed, single-stage, centrifugal units with vertical shafts.[26] The pump shaft is mounted on one radial bearing and one radial-thrust bearing. The distance between the axes of the impeller and the lower bearing is 2 m. The lower bearing does not operate in the sodium.[23] The pump can operate at two speeds and is driven by an electric motor. Characteristics of the primary-loop pumps are as follows:

Rated capacity, m^3/hr	3200
Rated head, m	110
Weight, metric tons	
Pump only	46.5
Pump and drive	86
Pump shaft speed, rpm	1000/250
Motor	
Type	Vertical, two-speed, squirrel-cage
Power at 1000 rpm, kw	1700
Power at 250 rpm, kw	55
Sodium height above impeller at start, mm	300

The Russians have a good deal of experience in the purification of sodium, specifically in limiting the oxygen content. It is anticipated[27] that, in the BN-350, even with the cold traps operating in an unfavorable mode, the oxygen purity in the primary circuit will not exceed 30 ppm. The primary circuit is provided with six cold traps. Apparently, the operation of the traps is not affected by the operation of a particular loop. One or two traps will normally be on at the same time,[23] but their cooling system is designed to handle four. Each trap, which is housed in a cylindrical vessel about 6 m high and 1 m in diameter, is designed for a sodium flow of 10 m^3/hr. A NaK system with two electromagnetic pumps (750 m^3/hr capacity each) and two air heat exchangers removes heat from the cold traps. A similar system cools the seals of the primary pumps.

Ten tanks, each of 50 m^3 capacity, are installed for draining the primary system.

INTERMEDIATE CIRCUIT

The sodium-cooled intermediate circuit transfers heat from the reactor system to the steam-water system. The circuit consists of six loops; as in the primary circuit, five are in use at rated power and one is in standby.

Sodium from the heat exchanger enters the superheater section of the steam generator at 453°C and leaves at 416°C. It then passes through the evaporator section of the steam generator, where its temperature is reduced to 273°C. The intermediate-circuit sodium is then pumped back to the heat exchanger for another pass.

The steam generators of the BN-350 have a recirculating boiler (evaporator) and a separate superheating section;[23-25] in comparison with the once-through designs generally specified elsewhere, the BN-350 units are large, complicated, and expensive. The units have undergone substantial design changes since the announcement of the reactor several years ago, but the design shown in Fig. 4 is thought to be the type that will actually be used in the plant. As the figure illustrates, each steam generator (there are six of them—one for each intermediate loop) consists of two parallel evaporator sections, two parallel superheater sections, and a surge tank. The sections are connected by long runs of pipe.

The natural-circulation boiler-evaporator uses bayonet tubes that are closed at the bottom end. Inside each tube is a concentric downcomer tube, and boiling takes place in the annulus between the downcomer tube and the bayonet tube. A single tube sheet holds the upper ends of the outer tubes, separating the sodium in the lower portion of the shell from the water and steam above. Single-wall separation is used. Above the tube sheet and the evaporator section is a space for natural separation of water and steam, the steam rising through what appears to be a demister section, then traveling through the pipe to the superheater section. The water is recirculated downward to the evaporator tubes, and feedwater is added by a header below the water level.

The superheater is a counterflow shell-and-tube heat exchanger, with sodium on the shell side and steam inside the single-wall tubes.

Thermal-expansion problems are minimized by the use of bayonet tubes in the boiling section, by making the superheater U-shaped, and by using long runs of pipe to connect the sections of the steam generator.

The surge tank is a cylindrical vessel that serves as part of the cover-gas system of the steam generator during normal operation. In the event of a sodium-water accident, the surge tank acts as a receiver to isolate the sodium that is ejected from the evaporator by gaseous reaction products. There is an emergency dump line for the water in the evaporators. Steam-generator characteristics are listed below.

Steam output, metric tons/hr	276
Superheated steam	
Pressure, kg/cm^2	50
Temperature, °C	435

Feedwater temperature, °C	158
Pressure in evaporator, kg/cm^2	53
Sodium temperature, °C	
At superheater inlet	453
At superheater outlet	416
At evaporator outlet	273
Sodium flow per loop, metric tons/hr	3260
Sodium pressure drop, kg/cm^2	
In superheater	0.47
In superheater/evaporating piping	0.14
In evaporator	0.32
Heat-transfer surface, m^2	
Evaporator	820
Superheater	455
Evaporator	
Diameter, mm	1400
Wall thickness, mm	24
Bayonet tubes	
Number	816
Spacing (triangular), mm	44
Diameter, mm	32
Wall thickness, mm	2
Evaporator bayonet tubes—*Continued*	
Inner (downcomer) tube diameter, mm	16
Wall thickness, mm	1.4
Superheater U-tubes	
Number	805
Diameter, mm	16
Wall thickness, mm	2
Spacing (triangular), mm	23
Material	2 Cr–Mo steel

The pumps in the intermediate circuit differ from those in the primary circuit only in the impeller and flow section. The technical characteristics of an intermediate-circuit pump are:

Rated capacity, m^3/hr	3700
Rated head, m	68
Weight, metric tons	
Pump only	25.5
Pump and drive	58
Pump shaft speed, rpm	1000/250
Maximum motor power, kw	1100

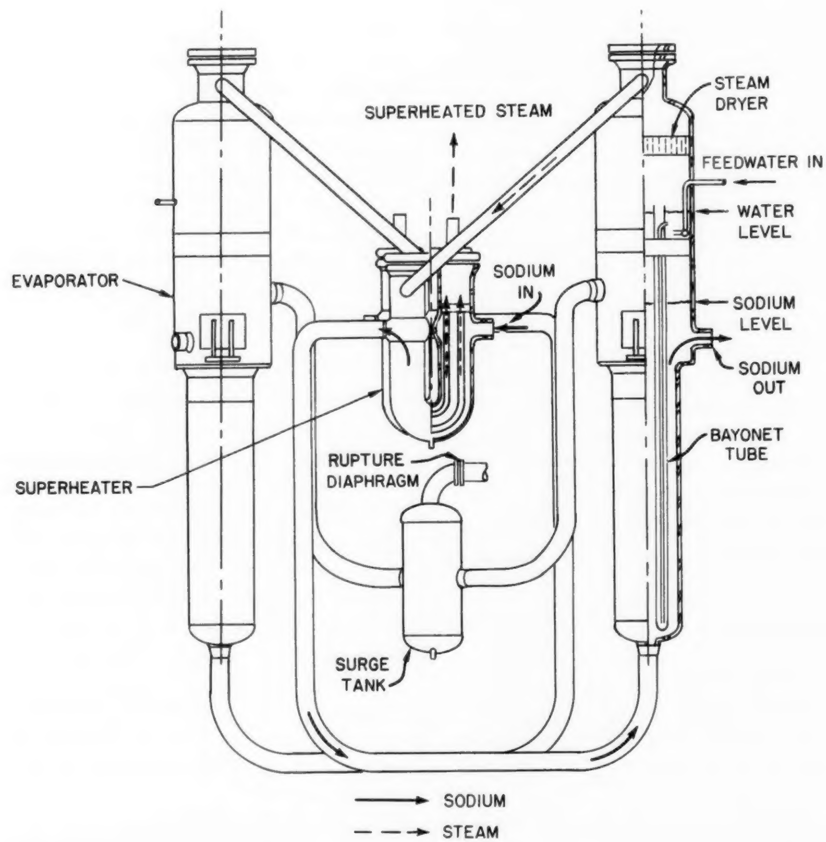

Fig. 4 Steam generator for the BN-350.

Each loop of the intermediate circuit has its own cold trap, identical in design to those of the primary circuit. The intermediate circuit also has four drain tanks, with a capacity of 50 m^3 each, which are designed to hold the sodium from any two of the six loops.

WATER–STEAM CIRCUIT

The steam produced in the steam generators is sent to three turbogenerators, each with a power of 50 Mw. After leaving the backpressure turbines, steam with a pressure of 5 to 6 atm is sent to the desalination complex where it is used to heat and evaporate seawater. As indicated in Fig. 1, condensate is returned from the desalination unit and passed through a heater which operates on exhaust or extraction steam from the turbine. The condensate then goes to a 6–atm deaerator, and feedwater is moved from the deaerator to the steam generators by a pump.

In the event of turbine failure, steam can be sent to the desalination complex through reduction cooling units which have a capacity of 250 metric tons/hr and which reduce the pressure from 50 to 5 or 6 atm. The reduction cooling units are utilized during reactor startup, when the steam conditions are such that it cannot be used in the turbines.

With regard to steam pressure, Ref. 9 contains the following statement: "The steam pressure in the steam generator is held constant at 50 atmospheres on all power levels. In order for the turbine to be able to receive the steam under these conditions, the pressure at the turbine input is lowered by a special reduction unit in conformity with the change in temperature. In this way, the turbine may operate over the entire indicated power range." The indicated power range is 30 to 100% of rated power.

A few details of the steam–water circuit are repeated below.

Steam	
Flow, metric tons/hr	1380
Pressure, kg/cm^2	50
Temperature, °C	435
Turbines	
Type	Backpressure
Number	3
Exhaust pressure, kg/cm^2	5 to 6
Generators	
Number	3
Output each, Mw	50
Feedwater temperature, °C	158

CONCLUSIONS

The BN–350 will likely be one of the world's first reasonably large, operating, fast breeder reactors. As such, it will furnish important data to its designers.

Intensive component–test programs have been carried out and should provide a measure of confidence that the plant will attain the modest goals set for it.

The reactor is far from an economically feasible one; it is a large, expensive experiment, despite the fact that it will produce steam for power production and water desalination.

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Sodium-Cooled Reactors: Current Status

By Myrna L. Steele*

Events related to the operation of plant systems in the period February–June 1969 are summarized below for EBR-II, the Fermi Fast Breeder Reactor, and Rapsodie. This summary updates the review article that appeared in *Reactor and Fuel-Processing Technology*, 12(3): 279-292.

EBR-II

As of June 20, 1969 (Run 35), the EBR-II had logged a total of 26,041 Mwd of successful operation.

Unscheduled Scrams. During the 50-kw tests at the beginning of Run 33A in March 1969 and the subsequent 50-Mw operations, six scrams occurred. One was caused by grounding in the fuel-element-rupture detection (FERD)-loop electronics; the other five were caused by an indicated high rate-of-change in the primary sodium flow. One of these five scrams was traced to a failure in the rate-of-change circuitry; the others were caused by actual flow perturbations from primary-pump control malfunctions.[1]

A spurious scram interrupted 50-Mw operation on April 18. After the primary cover gas was monitored for ^{133}Xe for 6 hr, power operation resumed and continued until erratic behavior of the FERD instrumentation began. Run 33 was terminated on April 22 because the instrumentation behavior was too unstable for continued operations.[2]

Secondary Sodium System. On March 13 an alarm indicated a sodium leak[1] past the first of two rupture disks in the north header of the secondary sodium system. The secondary system was drained, the leak probe was removed for inspection, and another rupture disk was installed.

Instrument-Air System. On Aug. 23, 1968, the exhaust valve of the No. 2 air compressor failed. Two piston-controlled check valves, specifically designed for pulsating-pressure service, were procured and installed at each of the outlet lines from the air compressors. As of May 1969,[2] these valves had operated satisfactorily for 1000 hr.

Primary Sodium System. A small sodium fire occurred during the sodium-sampling procedure in April.[2] The fire was confined to the room containing the cold trap in the purification system for primary sodium and was extinguished in about 4 hr. The fire was caused by a steady dripping of sodium from a failed tube in the electromagnetic pump. Damage occurred only to the sampling loop, and no personnel injuries or spread of contamination resulted.

FERD System. In May, during fuel handling for Run 34A, separate power supplies were installed for each FERD channel to mitigate the erratic behavior discussed above. Also, new cables were installed and rerouted to the cable-routing room.[2]

Reactor-Building Depressurization.[3] On March 28, the electrical solenoid on the exterior-air-supply valve in the reactor building failed, and the valve closed. The air-supply fan automatically tripped off; however, closure of the air-supply valve did not actuate the limit switch on the valve. The limit-switch actuation would have caused the building exhaust valve to close and prevent depressurization; however, the exhaust fans continued to run, thereby causing partial depressurization of about 0.5 psi in approximately 3 min. At the end of the 3-min period, partial isolation of the reactor building was manually initiated and the manual pressure-relief valve was opened to return the pressure in the building to normal. The failed solenoid was replaced, and reactor operation was resumed. During shutdown in the last week of May, the events of

*U. S. Atomic Energy Commission, Division of Technical Information Extension, Oak Ridge, Tenn.

March 28 were repeated. The system failure was immediately traced to the closed limit switch on the air-supply valve. The limit switch was adjusted.

FERMI FAST BREEDER REACTOR

Object Removal Program. In-place staking of the screw shanks on the zirconium flow guide was completed in January 1969.[4,7] The staking was done to prevent the shanks from becoming unscrewed during subsequent coolant flow. Vacuum-cleaning procedures were initiated to collect any debris from the arc-melting process. The outer shell of the arc-melt tool was converted to a vacuum-cleaner hose with the vacuum being provided by the waste-gas system; the gas removed from the primary system is replaced by clean gas from the holdup tank as it is required by the cover-gas pressure-control system. The total amount of debris collected by the vacuuming process amounted to about 1 oz. Repair of the primary and secondary piping that had been opened for object removal subsequent to the fuel-melting incident of 1966 was begun in February 1969;[5,7] difficulties in obtaining a satisfactory weld caused repairs to take longer than had been anticipated. Quality-control tests were completed on the 14-in. line in March.[6] Secondary containment, including insulation and induction heating, was restored. On March 22 the reactor vessel was filled with sodium and heated to 450°F for sodium cleanup.[6,8]

Steam Generators. All three steam generators were reassembled by February 1969.[4,7] Leak testing with nitrogen in the tube side showed that Nos. 2 and 3 appeared tight and that No. 1 had a tube-to-shell-side leakage of approximately 150 cm^3 of nitrogen per day at a pressure differential of 22 psig. Further leak testing is planned for No. 1. Tube-to-shell-side leaks in No. 1 were also reported in March 1969.[6,8]

Fuel-Handling System. An extensive fuel-handling program was completed in March[6,8] to make room in the core lattice for tools to inspect the hold-down mechanism. Difficulties were encountered because many positions had only one or two, rather than four, camming surfaces available. Efforts to move a safety rod from one position to another were discontinued when a safety rod appeared to stick in a guide tube.[6] The sticking was anticipated since oxide was suspected to have accumulated in the dashpot during draining of the primary system.

Primary Sodium Pumps. The fuel-handling program included refilling the primary system with sodium. After the refilling the three primary sodium pumps were run intermittently on pony-motor drive, and the No. 1 pump developed a torque such that it could not be rotated by hand with 400 ft-lb of applied force. In March, that pump was dismantled to determine the cause of the sticking and of the excessively high torque.

RAPSODIE

Operating Reliability. From the beginning of routine operation in August 1967 to the maintenance shutdown at the end of September 1968, Rapsodie logged 6777 hr of operation out of a possible 7425 hr. A total of 14 setbacks and 16 scrams occurred in the first year of operation, with 13 scrams being attributable to the on-line computer system, 1 to equipment failure, and 2 to human error.[9]

Fuel-Handling System. In late November 1967, rotation of the main rotating plug proved impossible. Inspection revealed that, although the method for cooling the plug was not adequate, the actual sticking was caused by accumulation of sodium and sodium oxide. Using the nitrogen cooling circuits to heat the plugs relieved the immediate sticking; later, plug-cooling procedures were modified to provide for frequent rotations during plug cooling. The plugs also were operated at higher temperatures.[9]

In a second incident, Dec. 2, 1967, a misalignment of the hold-down tube damaged the bottom of the tube and caused a blanket-fuel-subassembly head to become twisted. This deformation was later responsible for the partial withdrawal of a reflector subassembly. To prevent such incidents in the future, workers modified the bottom of the hold-down tube to assure that it centered properly when sliding down on the head of the subassemblies and to limit the forces exerted on the subassembly heads.

A third incident involved the inability of the fuel-handling machine to lift a control rod because the gripper jaws had become twisted. A thermal-gradient-induced torque is applied constantly to the fuel-handling machine as it is used. During the first few months of Rapsodie's operation, the opening and closing motion of the gripper jaws was motor controlled. The authors of Ref. 9 propose that it is possible that a malfunction of the torque limiter could have allowed excessive force to be applied to the gripper, causing it to twist gradually. The motor drives on the gripper have been replaced.[9]

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ECONOMICS

Economics of Plutonium Recycle in Thermal Reactors

By Henry D. Raleigh*

A survey[1] of U. S. utilities on fuel supply for 83 nuclear plants, in operation, under construction, or on order, with a total capacity of 61,800 Mw(e) indicates that the supply of uranium is adequate for purchase commitments to 1982. However, the possibility of using plutonium as the initial fuel enrichment in thermal power reactors is being seriously investigated by both government and industry in the United States and other countries. The principal reasons for these studies in the United States are the 1971 termination of the government buy-back of plutonium produced in commercial power reactors, the anticipated timing in the wide use of commercial fast breeders fueled with plutonium, and an immediate economic need by power-reactor operators to find a market for produced plutonium. A significant economic benefit to nations without uranium-enriching facilities is that by plutonium recycle they may no longer need to rely completely on imported uranium as their only enriched-fuel supply.

In September 1956 the U. S. Atomic Energy Commission (AEC) established the Plutonium Utilization Program[2] to investigate the technical and economic aspects of plutonium recycle. The program, which is now being phased out,[3] has been conducted primarily by General Electric Company at Hanford Laboratories and by Battelle Memorial Institute at Pacific Northwest Laboratory. Information on the neutronics aspects of plutonium recycle, much of it provided by the Plutonium Utilization Program, appears in a previous issue of this journal.[4] As the result of recommendations included in a 1964 survey on plutonium fuels technology, the Edison Electric Institute is continuing studies to evaluate the technical and economic aspects of utilizing plutonium in light-water reactors.[5]

PLUTONIUM SUPPLY AND DEMAND

Projections of plutonium availability depend primarily on the projection of installed nuclear plant capacity as a function of time. Projections of installed plant capacity vary considerably, particularly in the period following 1975.

A report on the outlook for U. S. light-water reactors[6] indicates about 135,000 kg of fissile plutonium recovered by 1981. The cumulative need for plutonium for activities related to the development of fast breeders is indicated as about 10,000 kg. Plutonium availability projections are shown in Fig. 1.

Another projection[7] shows that 100 to 160 metric tons of plutonium will be recovered by 1980 and 330 to 480 metric tons recovered by 1985. About 65 to 70% of these totals represents fissile plutonium. A plutonium excess in the United States is predicted[8] for around 1973. Fast reactors are not expected to be introduced until 1980, and substantial plutonium stockpiling will not take place until the mid-1980's. The AEC plans[9] to make immediate use of the plutonium received from utilities in its research and development programs. It will not be stockpiled for future distribution. The total investment in plutonium purchased by Dec. 31, 1970, when the guaranteed buy-back price of $10 per gram of contained fissile isotopes is terminated, is expected to be $80 to $100 million. World production of fissile plutonium is

*U. S. Atomic Energy Commission, Division of Technical Information Extension, Oak Ridge, Tenn.

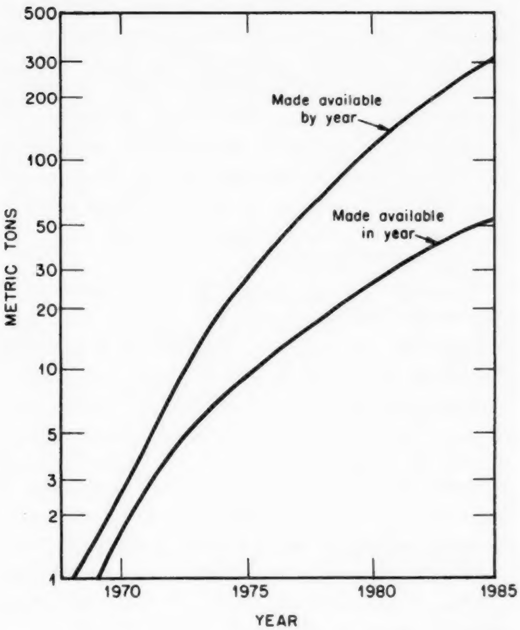

Fig. 1 Metric tons of fissile plutonium from light-water reactors.

predicted[10] to rise from a range of 3 to 5 metric tons per year in the early 1970's to 25 metric tons per year in the early 1980's. Total fissile plutonium produced by the mid-1980's will be 200 to 400 metric tons. The world demand for plutonium[2,11] for fast breeder development should be fully covered by 1975. Projections of the minimum needs of plutonium for the long-term reactor development programs of the German Federal Republic[11] and Japan[12] indicate that plutonium must be bought from other nations.

STORAGE

In the United Kingdom, France, and Russia, plutonium is largely reserved[13,14] for fast reactor programs because inventory charges for government holdings of plutonium can be set quite low. The United Kingdom fast reactor effort is planned to be commercial by 1971 and will provide more economic utilization of plutonium than thermal recycle. The Canadian program[15] for plutonium utilization is to store the spent heavy-water-reactor fuels until the plutonium can be sold at a profit. No value has been placed on it in calculating fuel-cycle costs. As large fuel-reprocessing plants come into operation and plutonium-extraction costs drop, recycle in heavy-water-reactor systems will be considered. It is expected that for several years other users will be prepared to buy the plutonium at a greater value than can be obtained by recycling in CANDU-type plants.

If plutonium recoverable from U. S. light-water reactors were assigned a zero value, the fuel-cycle cost would be increased[2,7,8,16] 0.2 to 0.4 mill/kw-hr, or $1.5 to $3 million per year per 1000-Mw(e) reactor. A central aspect in the evaluation of plutonium-recycle economics is the so-called "value." Value is defined as the maximum price that can be assigned to plutonium in a given reactor to produce the same unit power costs whether the reactor is fueled with plutonium or enriched uranium. If no market exists for plutonium at the assumed price and it must be carried on inventory, the capital outlay for plutonium would double every 7 years.[17] Assuming that a typical 500-Mw(e) plant produces about a million dollars' worth of plutonium a year, the capital-cost factor will become significant if reactor use of plutonium is long delayed. This is illustrated in Table 1. The plutonium discharged is valued at $10 per gram.

If a utility decides to store plutonium in anticipation of a future price rise, one of the major economic factors is actual storage cost. Storage costs at AEC facilities are estimated[7] at $0.20 per gram of fissile plutonium per year. Commercial-facilities storage costs would be $0.35 per gram. The increase in plutonium price due to potential increases in the cost of enriched uranium or reduction of the plutonium fabrication-cost penalty will not justify plutonium

Table 1 Approximate Capital Value of Accumulated Plutonium Inventory from a 500-Mw(e) Nuclear Power Plant Starting Operation in 1970

Year	Value of plutonium discharged	Capital charges at 6.9% (after taxes) compounded	Cumulative cost of plutonium inventory
1973	$1,000,000	$	$ 1,000,000
1974	1,000,000	67,000	2,067,000
1975	1,000,000	139,000	3,106,000
1976	1,000,000	210,000	4,316,000
1977	1,000,000	289,000	5,605,000
1978	1,000,000	376,000	6,981,000
1979	1,000,000	467,000	8,448,000
1980	1,000,000	565,000	10,013,000
1981	1,000,000	673,000	11,686,000
1982	1,000,000	783,000	13,469,000
1983	1,000,000	904,000	15,373,000

storage for more than 2 or 3 years. Even the estimated highest probable rise in market price of plutonium to 50% greater than the price of ^{235}U in 90% enriched uranium cannot justify a storage period longer than 5 years.

PHYSICS AND FABRICATION COSTS

Although countries other than the United States anticipate little need for plutonium recycle, a number of them are studying aspects of plutonium utilization. The European Atomic Energy Community research facilities are conducting[18] programs on plutonium fuel fabrication and processing. Experimental plutonium-bearing fuel elements are being irradiated in several thermal power reactors owned by members of Euratom. Belgium has pursued[19] an experimental program on plutonium fuel since 1960. The United Kingdom has studied[13] fabrication processes and costs for plutonium-enriched fuel and expects no major problems with fueling AGR-type reactors with plutonium. The minimum market price is forecast[10] to be in the range \$7 to \$12 per fissile gram in the early 1970's. One analysis[11] indicates thermal recycle will be economic only at prices below \$5 per gram. Assuming the market price of plutonium in the mid-1980's will increase to over \$36 per gram, Britain anticipates a savings of over \$700 million in the next 30 years by stockpiling plutonium for use in fast breeders.

The effect of plutonium enrichment on unit fueling costs of Canadian heavy-water reactors has been calculated.[20] An enrichment of 4.48 g of fissile plutonium per kilogram of uranium shows a distinct advantage over natural-uranium fuel. At this enrichment the break-even cost of plutonium is \$12.90 per gram. The minimum economic size of a fuel-processing and recovery plant will require a throughput of more than 200 metric tons of fuel per year. Since this situation will not exist in Canada until the mid-1970's, Canadian fuel will most probably be processed on a contract basis in other countries. Estimates of processing costs in the United Kingdom in the 1970's indicate that the fissile-plutonium recovery cost could be as low as \$2.00 per gram. Even at a price of \$9.00 per gram, plutonium recycling will reduce the heavy-water-reactor fuel-cycle cost by 10%.

Initial introduction of plutonium into light-water reactors will require reoptimization of uranium lattices for plutonium.[21] The plutonium fuels burn up more rapidly than ^{235}U fuels mainly because the ^{239}Pu thermal-neutron cross section is about twice that of ^{235}U. When plutonium fuel assemblies are placed next to uranium fuel assemblies of the same fissile enrichment, the specific power will be higher in the plutonium fuel. This creates serious power-peaking problems. The most significant factors in lattice reoptimization for water reactors are plutonium fuel composition, lattice pitch and rod size, and spectral softening. Plutonium recycling in water reactors will change the neutron spectrum so that a slight increase in the moderator-to-fuel ratio could decrease the fuel-cycle cost[22] by about 0.1 to 0.2 mill/kw-hr.

Most of the U. S. studies on lattice optimization have been conducted at the Pacific Northwest Laboratory (formerly Hanford Laboratories). Advantages and disadvantages of varying fuel density, lattice spacing, and fuel-rod diameter were calculated[23] for a PuO_2-UO_2 fueled optimized lattice in a pressurized-water reactor similar to Yankee. Several different variations of lattice parameters resulted in lower fuel-cycle costs than the 1.84 mills/kw-hr reference uranium lattice. The Yankee reactor has been estimated[24] to be deriving 20% of its power from plutonium produced in the fuel with no serious operating problems. Plutonium values of \$10 to \$13 per gram were calculated[25] for the first recycle step in three successive recyclings for five types of reactors (pressurized water, boiling water, organic moderated, heavy-water moderated, and gas cooled). Recovery and use of plutonium at these values is worth 0.3 to 0.5 mill/kw-hr.

If the fabrication costs of plutonium and uranium fuels are assumed to be equal, plutonium is worth \$7500 per kilogram as a boiling-water reactor fuel based on an ore cost[26] of \$5.75 per pound of U_3O_8. However, cost disadvantages ranging from 20 to 40% are predicted[8] for processing plutonium fuels as compared to uranium fuels. The factors most influencing plutonium fabrication cost are plutonium toxicity and fabrication-plant capacity. For equivalent plant throughput, one analysis[8] indicates fabrication costs will be 20 to 25% higher for plutonium than uranium. This fabrication cost penalty is illustrated by Fig. 2.

A fabrication-plant throughput which may represent the early 1970's is a quarter ton of plutonium fuels per day and 2 tons of uranium fuels per day. This throughput differential, as shown by Fig. 2, results in an incremental fabrication cost for plutonium of \$35 per kilogram over the uranium fabrication cost.

A cost breakdown for a 1000-kg order indicates[27] the importance of processing costs. Based on the price of plutonium purchased from the AEC at \$43 per gram

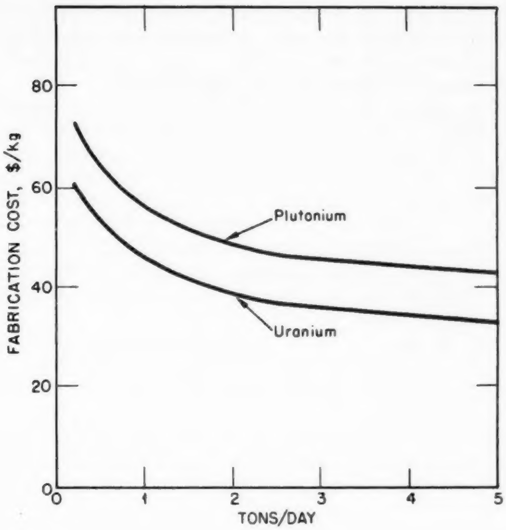

Fig. 2 Fabrication of plutonium-bearing fuel vs. uranium-only fuels.

of fissile isotopes, the fabrication cost of fuel elements containing more than 5% plutonium would be approximately $5000 per kilogram of contained plutonium. For a 1000-kg order and using plants of the size dictated by anticipated plutonium requirements, the cost to the customer would be $5 million. Because of industry immaturity, such items as analytical chemistry, quality control, and shipping present significant cost problems. Table 2 shows the cost breakdown for the 1000-kg order.

Table 2 Cost Breakdown for Fabrication of Fuel Elements Containing 1000 kg of Plutonium

	Millions of dollars	Percent of total
Processing and fabrication	1.20	24
Analytical chemistry and quality control	0.62	12
Use charges (500 kg over 1 year)	0.95	19
Insurance	0.15	3
Withdrawal, handling, analyses, and shipping charges	0.40	8
Shipper–receiver difference	0.08	2
Processing losses at 2%	0.80	16
Scrap recovery and recycle	0.80	16
Total cost	5.00	100

One study has determined[28] that water reactors with close-coupled reprocessing and fabrication facilities may offer economic advantages over central-plant reprocessing.

UTILIZATION

A plutonium self-sustaining cycle was evaluated[29] for a pressure-tube-type heavy-water reactor. The cycle starts with natural-uranium fuel. Natural uranium is the base fuel in subsequent cycles with all the plutonium from the previous cycle included. The self-sustaining cycle is equivalent to the use of about 1% enriched uranium.

Phoenix fuel is plutonium with a relatively high ^{240}Pu content. Neutron capture in the ^{240}Pu produces fissile ^{241}Pu, which replaces some of the ^{239}Pu lost from the system. A study[30] of fuel-cycle costs in the Consolidated Nuclear Steam Generator concept, as developed by Babcock & Wilcox for merchant-ship propulsion, shows Phoenix fuels to be competitive with slightly enriched-uranium fuel. Fuel-cycle costs ranged upward from 2.54 mills/shp-hr for various Phoenix-fueled systems. The fuel-cycle cost for the enriched-uranium base case was 2.95 mills/shp-hr.

A model of the U. S. electric-power economy assuming a mixed system for a light-water breeder reactor indicates[8] that the value of plutonium will rise from $8 per gram in the early 1970's to $15 per gram by 1980. The minimum plutonium value required for recycle is $1 to $3 per gram. Only light-water reactors will be operating in the 1970's, and 20, out of a total of 141, will be involved in plutonium-recycle operation. The breeders introduced in 1980 are predicted to have such a high doubling time that their owners could not afford to pay the $6 to $10 value for plutonium in thermal recycle. The breeders introduced by 1994 are expected to have doubling times good enough to cause significant stockpiling of plutonium a few years prior to their introduction.

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Correction

Joseph W. Ray, the author of the article Tritium in Power Reactors (Vol. 12, No. 1, pages 19–26), points out that in his Eq. 2 on page 20 the constant should be 8.95×10^{10} and *not* 9.85×10^5. (Mr. Ray notes that the curves in Fig. 1, which are partly based on Eq. 2, are correct as shown, since the correct constant was used in the calculations for the figure.)

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